Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Residual Heat Removal	Group #	1		
	K/A #	005 K2.03		
	Importance Rating	2.7		
Knowledge of bus power supplies to the following: RCS pressure boundary motor-operated valves				

Question # 1

What is the power supply to the 'A' RHR Loop 1 Inlet Isolation Valve, EJHV8701A?

A. NK01

B. NK02

C. NG01

D. NG02

Answer: C

Explanation:
RHR Train A (B) is connected to the RCS Loop 1 (4) hot leg through RHR Pump Suction Valves (aka or also transferred to as the Loop Isolation Valves) BBPV8702A (B) and EJHV8701A (B). These valves are motor operated valves located inside containment and are controlled from Main Control Board Panel RL017. BBPV8702A and BBPV8702B are powered from NG02B. EJHV8701A and EJHV8701B are powered from NG01B. (Notice the cross train power supplies for example BBPV8702A is powered from NG02 A valve from a B safety related bus. This adds to the plausibility of opposite train power supplies).
NK01 and NK02 are 125 VDC safety related busses which power some RCS isolation valves and therefore are plausible.

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Correct – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):
1. E-23EJ05A(Q), Schematic Diagram RHR Loop 1 Inlet Isolation Valve, Rev 9

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Learning Objective: T61.0110, systems, LP #7, Objective B: DESCRIBE the purpose and operation of the following RHR System components, to include interlocks, controller operation and power supplies.
3. Reactor Coolant System (RCS) Hot Leg Suction Valves to RHR

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
	Group #	1		
Large Break LOCA / 3	**K/A #**	000011 EK2.02		
	Importance Rating	2.6		
Knowledge of the interrelations between the Large Break Loca and the following: Pumps.				

Question # 2

Reactor Power was 100% power when a LOCA occurred.

The crew has just transitioned from E-1 to ES-1.2, Post LOCA Cooldown and Depressurization, and currently performing step 1 of ES-1.2.

Current plant conditions are as follows:

Core Exit Thermocouples	375°F - Lowering
Wide Range Thot	425°F - Lowering
RCS Pressure	400 PSIG and Stable
Subcooling Meter	3°F -Subcooled
Containment Pressure	28 PSIG - Lowering
Containment Temperature	175°F - Stable
Containment Radiation	5 x 10^3 R/HR - Stable
RWST Level	87% - Lowering
Secondary Radiation Levels	Normal
S/G Levels	3.6% NR rising
S/G Pressures	Stable

What is the status of the Reactor Coolant and RHR Pumps?

	RCPs Running	RHR Pumps Running
A.	NO	NO
B.	NO	YES
C.	YES	NO
D.	YES	YES

Answer: A

Explanation: *Based on the given conditions, RCS pressure less than 1425 psig and Containment pressure greater than 27 psig, RCP should be secured. This is covered by both foldout page criteria and a step in E-0 prior to the transition to E-1. Based on the conditions given RHR pumps are secured due to SI pumps running and maintaining RCS pressure greater than 325 psig. Note: E-1 also contains the same foldout page criteria as E-0 with respect to tripping RCPs.*

A. Correct – See above
B. Incorrect - See above
C. Incorrect – See above
D. Incorrect – See above

Technical Reference(s):
1. E-0 Reactor Trip or Safety Injection, Rev 20
2. E-1 Loss of Reactor or Secondary Coolant, Rev 19

References to be provided to applicants during examination: None

Learning Objective: T61.003D LP#8 Obj I. STATE and EXPLAIN the parameters which are evaluated, including their Criteria and Basis, to transition from E-1 to other procedures.

Question Source: Bank # __R11859____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2014 NRC ILT Exam Q#45_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(8)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
(EPE 7; BW E02&E10; CE E02) Reactor Trip	**Group #**	1		
	K/A #	007 EK1.05		
	Importance Rating	3.3		
Knowledge of the operational implications of the following concepts as they apply to reactor trip: Decay Power as a function of time				

Question # 3

Following a Reactor trip from 100%, what is the MINIMUM time that it takes reactor power to decay before Source Range NI's will energize?

A. 5-7 minutes

B. 8-11 minutes

C. 12-15 minutes

D. 16-18 minutes

Answer: C

Explanation:
The reactor will take ~12-15 minutes to transition to the source range and at that point SR NI will be energized (4-5 decades at -1/3 DPM SUR)

A. Incorrect - SR NI will not be energized
B. Incorrect - SR NI will not be energized
C. Correct – SR NI will be energized
D. Incorrect - SR NI will already be energized

Technical Reference(s):
1. Nuclear Reactor Theory, Lamarsh, September 1972
2. E-0 Reactor Trip or Safety Injection, Rev 20

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #28, Objective D: IDENTIFY the Nuclear Instrumentation System Main Control Board and NIS Panel controls, alarms and indications and DESCRIBE how each is used to predict, monitor and control the Nuclear Instrumentation System.

T61.GFES, LP #44, Objective #23 - #25:
23. Explain the shape of the curve of reactor power versus time after a reactor trip.

24. Define decay heat.

25. Explain the relationship between decay heat generation and:
  a. Power level history
  b. Power production
  c. Time since reactor shutdown

Question Source: Bank # ______
Modified Bank # _____
New __X_____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(1)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	1		
Small Break LOCA	**Group**	1		
	K/A	009 EA1.15		
	Importance Rating	3.9		
Ability to operate and monitor as they apply to a SBLOCA: PORV and PORV block valve				

Question # 4

Following a Small Break LOCA, the crew has just entered ES-1.2, Post LOCA Cooldown and Depressurization, with the following plant conditions:

- Hot leg temperatures:
  - Loop A 470°F
  - Loop B 320°F
  - Loop C 480°F
  - Loop D 440°F
- Reactor head temperature 564°F
- Highest core exit thermocouple (CET) 512°F
- PZR Vapor temperature is 585°F
- RCS Pressure 1385 psig
- All SI and Charging pumps are running

The RO opens a PORV to lower RCS pressure.

Based on these plant conditions, indications of voiding in the Reactor Coolant System ….

A. will first be seen at approximately 551 psig

B. will first be seen at approximately 743 psig

C. will first be seen at approximately 1155 psig

D. will first be seen at approximately 1377 psig

Answer: C

Explanation:

A: INCORRECT: This distractor is credible because it corresponds with the highest hot leg temperature, however, voiding will first occur when pressure reaches saturation for the hottest spot in the RCS (head)
B:. INCORRECT: This distractor is credible because it corresponds with the saturation pressure for the highest reading CET, however, voiding will first occur when pressure reaches saturation

for the hottest spot in the RCS (head).
C: CORRECT: Voiding will occur when pressure reaches saturation for the highest temperature in the RCS which corresponds to the given temperature of the Reactor Vessel Head
D: INCORRECT: This distractor is credible because it corresponds to the saturation pressure for PZR Vapor temperature

Technical Reference(s):
1. ES-1.2, Post LOCA Cooldown and Depressurization, Rev 18

References to be provided to applicants during examination: None

Learning Objective: T61.003D, emergency Operations, LP #10, Objective I &J:
I: OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of ES-1.2, Post LOCA Cooldown and Depressurization.

J: STATE the Subcooling requirements as defined in ES-1.2.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Reactor Coolant Pump Malfunctions	Group #	1		
	K/A #	00015 AA2.09		
	Importance Rating	3.4		
Ability to determine and interpret the following as they apply to RCP malfunctions: When to secure RCPs on high stator temperatures				

Question # 5

Reactor Power is 45%.

The crew is performing OTO-BB-00002, RCP Off-Normal, due to the following Reactor Coolant Pump (RCP) 'A' parameters:

- Thrust Bearing Upper Temperature = 189°F
- Motor Stator Winding Temperature = 318°F

(1) What temperature limit has been exceeded?

And

(2) Per OTO-BB-00002, what action should the crew take NEXT?

A. (1) Motor Stator Winding Temperature
(2) Trip RCP 'A' ONLY

B. (1) Motor Stator Winding Temperature
(2) Immediately trip the Reactor and then trip RCP 'A'

C. (1) Thrust Bearing Upper Temperature
(2) Trip RCP 'A' ONLY

D. (1) Thrust Bearing Upper Temperature
(2) Immediately trip the Reactor and then trip RCP 'A'

Answer: A

Explanation:
Per OTO-BB-00002, continuous action step #C1, the motor stator winding temperature has exceeded its limit of 311F and the RNO therefore applies. The Motor Bearing Temperature is under is procedural limit.

As Reactor Power is less than P-8 (48%) and no other mention of B through D RCPs are affected the correct action is the Trip the A RCP ONLY. Tripping the reactor and then the 'A' RCP is plausible if the candidate does not correctly process than the reactor is below P-8 or does not recall and apply the OTO procedure steps correctly for multiple RCPS affected between the power levels of 10% to 48%.

A. Correct – See above explanation
B. Incorrect – See above explanation. Part 2 is wrong
C. Incorrect – See above explanation. Part 1 is wrong
D. Incorrect – See above explanation. Both parts are wrong

Technical Reference(s):
1. OTO-BB-00002, RCP Off-Normal, Rev 33

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #11, Objective C, D, G:
C: DESCRIBE Continuous Action Step(s) including the required Response Not Obtained actions.

D: Given a set of plant conditions or parameters indicating a RCP Off-Normal condition, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

G. Using OTO-BB-00002 RECOGNIZE the conditions that would require a Reactor Trip/Turbine Trip.

Question Source: Bank # ______
Modified Bank # ___X – Bank question id#L16778___
New _______

Question History: Last NRC Exam ___L16778 was last used on the 2011 ILT NRC exam____

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Modified the Power level and temperatures from the bank question in effect changing both the answers for part 1 and part 2 of the question

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	1		
Loss of Reactor Coolant Makeup	Group	1		
	K/A	022 AA2.04		
	Importance Rating	2.9		
Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: How long PZR level can be maintained within limits				

Question # 6

OTO-BB-00003, Reactor Coolant System Excessive Leakage, has been entered due to RCS leakage of 8 gpm.

- PZR Level is 34%
- Tave is constant
- Letdown is stable at 120 gpm
- Charging is in manual and stable at 132 gpm

What is the MAXIMUM amount of time before letdown isolates with NO OPERATOR ACTION?

A. 43 minutes

B. 68 minutes

C. 128 minutes

D. 255 minutes

Answer: C

Explanation:
The limit in question is 17% in which letdown isolates and PZR heaters turn off

34%-17% (Letdown isolates) = 17%
17% x 60 gal/% = 1,020 gal
132 gpm charging - 120 gpm letdown - 12 gpm seal leak off - 8 gpm RCS leakage = -8 gpm lost from PZR

1,020 gal / 8 gpm = 127.5 minutes ~ 128 minutes

If student uses 25% for letdown isolation = 67.5 minutes ~ 68 minutes
If student uses VCT 20 gal/% = 42.5 minutes ~ 43 minutes
If student uses cold cal PZR 120 gal/% = 255 minutes

A: INCORRECT: see above

B: INCORRECT: see above
C: CORRECT
D: INCORRECT: see above

Technical Reference(s):
1. M-22BG01(Q), P&ID CVCS System, Rev 33

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #11, Objective B: DESCRIBE the purpose, operation and interlocks for the following CVCS components:
1. Letdown Isolation Valves

Question Source: Bank # __16383____
Modified Bank # ___
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of RHR System / 4	Group #	1		
	K/A #	00025 G2.1.23		
	Importance Rating	4.3		
Ability to perform specific system and integrated plant procedures during all modes of operation				

Question # 7

The Plant is in MODE 5.

- Reduced Inventory Operations are in progress for "B" RCP repair.
- RHR Pump "A" and CCW Train "A" are in service.

Then:

- RHR Pump "A" discharge flow begins fluctuating between 1500-1800 gpm
- RCS temperature is rising slowly
- RCS level indicates the following:
    - Loop 1 Hot Leg Level indicates 15.5 inches
    - Loop 4 Hot Leg Level indicates 15.0 inches

The crew enters OTO-EJ-00003, Loss of RHR While operating at Reduced Inventory or Mid-Loop Conditions.

Per OTO-EJ-00003, what is the FIRST action that should be taken?

A. Reduce RHR flow to <1350 gpm

B. Isolate Letdown and known Drain Paths

C. Initiate RCS makeup using an available SI Pump

D. Stop RHR Pump(s) and place in Pull To Lock (PTL)

Answer: D

Explanation:

Based on RCS level given in the stem at step #2b, the RNO will be implemented which is to stop the RHR pumps and place in PTL.

A. Incorrect – If Level was greater than 16 inches, then this action in step #2c of the OTO would be correct.
B. Incorrect – this is an appropriate action but not the first action as this is step #3 of the OTO.
C. Incorrect – This is an action in step #8 RNO and as the stem says that RCS temperature is rising, this is an plausible FIRST action to take however this is not the FIRST action per the procedure for the crew to implement.
D. Correct – see above explanation

Technical Reference(s):
1. OTO-EJ-0003, Loss of RHR While Operating at reduced Inventory or Mid-Loop Conditions, Rev 11

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #62, Objective H - STATE major action categories and symptoms/entry conditions for OTO-EJ-00003, Loss of RHR while operating at reduced inventory or mid loop.

Question Source: Bank # ______
Modified Bank # __ X L16500____
New _______

Question History: Last NRC Exam _____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Modified stem and correct answer and changed one distractor to "Isolate Letdown and Known Drain Paths"

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier	1		
Loss of Component Cooling Water	**Group**	1		
	K/A	026 AA1.02		
	Importance Rating	3.2		
Ability to operate and/or monitor the following as they apply to loss of CCW: Loads on the CCWS in the control room				

Question # 8

Reactor Power is 100%.

- "B" Component Cooling Water (CCW) train is running supplying the Service Loop
- "B" CCW surge tank is at 8% and lowering
- "A" CCW surge tank indicates 55% and stable

What is the NEXT required action in accordance with OTO-EG-00001, CCW System Malfunction, while attempting to identify and isolate the leak?

A. Align ESW makeup to the 'B' CCW train

B. Fill Surge tank by opening EGLV0001, DI to surge tank

C. Trip the reactor and trip RCPs, then perform E-0, Reactor Trip Or Safety Injection

D. Transfer the Service Loop to the “A” Train while maintaining reactor power stable

Answer: C

Explanation:

A: INCORRECT: Step 9 RNO but don't perform since no emergency conditions exists .
B: INCORRECT: Step 9 RNO for A surge tank
C: CORRECT: Step 10 RNO since B Surge Tank level is not greater than 10%
D: INCORRECT: Step 12 directs Attachment B which realigns Service loop

Technical Reference(s):
1. OTO-EG-00001, CCW System Malfunction, Rev 16

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Learning Objective: T61.003B, Off Normal Operations, LP #17 Objective E: Given a set of plant conditions or parameters indicating a CCW System Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # __ X__L16502___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	1		
Pressurizer Pressure Control System Malfunction	Group	1		
	K/A	027 AA2.04		
	Importance Rating	3.7		
Ability to determine and interpret the following as they apply to the PZR Pressure control malfunction and the following: Tech- Spec Limit for RCS Pressure				

Question # 9

Reactor Power is 100%.

A malfunction in the Pressurizer Pressure Control System is causing Pressurizer pressure to lower.

As pressure lowers, what is the FIRST pressure in which a Technical Specification is NOT met?

A. 2218 psig

B. 2210 psig

C. 2193 psig

D. 1970 psig

Answer: C

Explanation:
A INCORRECT: 2218 psig is plausible because it is a pressure associated with the Pressurizer pressure control systems (the pressure at which backup heaters will turn off if they were on).
B: INCORRECT: : 2210 psig is plausible because it is a pressure associated with the Pressurizer pressure control systems (the pressure at which backup heaters will turn on).
C: CORRECT: The DNB TS 3.4.1 wil not be met if pressure lowers below this COLR value in mode 1. To avoid T.S. entry,pressure must be maintained >2195 psig.
D: INCORRECT: Plausible but incorrect as this isn't the FIRST TS not met on the pressure reduction. Per TS 3.3.2 EFSAS Interlocks P-11 (function 8.b) is not met below the listed value.

Technical Reference(s):
1. TS 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
2. COLR UL NRC06413, page 21
3. OTN-BB-00005, Pressurizer and Pressurizer Pressure Control, Rev 15 Attachment 1
4. TS 3.3.2, EFSAS Instrumentation

References to be provided to applicants during examination: None

Learning Objective: T61.0110 systems, LP #9, Objective M: STATE the LCOs for the RCS Pressure and Temperature Limit Technical Specifications and IDENTIFY the limiting components for the RCS heatup and cooldown curves, and EXPLAIN the basis.

Question Source: Bank #______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Anticipated Transient Without Scram / SF1	**Group #**	1		
	K/A #	029 G2.1.8		
	Importance Rating	3.4		
Ability to coordinate personnel activities outside the control room				

Question # 10

Reactor Power was 100% when a reactor trip signal was generated.

- The Reactor failed to trip and the crew performed the immediate actions of E-0, Reactor Trip or Safety Injection
- FR-S.1, Response to Nuclear Power Generation, has been entered
- The crew has just completed step #5, 'Check Containment Purge isolation', and is proceeding to step #6 'Check if the following trips have occurred'

Per FR-S.1, what action(s) should the control room direct the Primary OT to perform FIRST?

A. Locally trip the reactor trip and bypass breakers

B. Locally open the supply breakers to PG19 and PG20

C. Locally place the MG set motor circuit breaker control switches in Pull-To-Lock

D. Locally place the MG set generator circuit breaker control switches in Pull-To-Lock

Answer: A

Explanation:

A. Correct. The first action per FR-S.1, Step 6 RNO a. Locally Trip Reactor at the Reactor Trip Switchgear by tripping the Reactor Trip and Bypass Trip Breakers.
B. Incorrect – plausible as the action to open the PG19 and PG20 supply breakers is a RNO action of E-0 immediate actions (and a recent E-0 procedure update). While this action would have been done in E-0 by the Control Room Personal, FR-S.1 does not direct opening these supply breakers. The MG sets are powered from PG19 and PG20 and this action in effect deenergizes the MG Sets from the control room but are not local field actions per the procedures which makes it a plausible way to insert negative reactivity from in plant actions but incorrect.
C. Incorrect. FR-S.1 Step 6 RNO if local Reactor Trip and Bypass Trip Breakers will not open, then place Motor Circuit breaker control switches in PULL-TO-LOCK and therefore is plausible as it is a procedure step but incorrect as it is not the first in field action directed per FR-S.1.

D. Incorrect. FR-S.1 Step 6 RNO if local Reactor Trip and Bypass Trip Breakers will not open, then place Generator Circuit breaker control switches in PULL-TO-LOCK and therefore is plausible as it is a procedure step but incorrect as it is not the first in field action directed per FR-S.1.

Technical Reference(s):
1. FR-S.1, Response to Nuclear Generation, Rev 12
2. E-0, Reactor Trip or Safety Injection, Rev 20

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP#29, Objective K: Outline procedureal flowpath including major system and equipment operation in accomplishing the goal of FR-S.1, Response to Nuclear Power Generation / ATWS.

Question Source: Bank # __X L7654____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	1		
Steam Line Rupture—Excessive Heat Transfer	**Group #**	1		
	K/A #	040 AK3.04		
	Importance Rating	4.5		
Knowledge of the reason for the following responses as they apply Steam line rupture: Actions contained in EOPS for steam line rupture				

Question # 11

What is the BASIS for isolating all feedwater to a faulted Steam Generator (SG) per E-2, Faulted SG Isolation?

A. To minimize RCS cooldown and mass/energy release following a steam line break

B. To ensure the release to the environment remains below the 10CFR100 limits on a design basis event

C. To reduce the probability of occurrence of a steam generator tube rupture in the faulted steam generator

D. To prevent all feedwater flow from entering the faulted steam generator and filling the generator, causing the steam generator atmospheric dump valve to actuate to lift

Answer: A

Explanation:
Per the basis document for E-2, step #4, Isolation of the feedwater to the faulted SG maximizes the cooldown capability of the nonfaulted loops following a feedline break and minimizes the RCS cooldown and mass and energy release following a steamline break. Isolation of steam paths from the faulted SG also minimizes the RCS cooldown and mass and energy release to containment. In addition, isolation of these steam paths could isolate the break.

A. Correct – See explanation above
B. Incorrect – this is one of the reasons for E-1 step #16 "If fuel damage has occurred following a LOCA, there is a potential for releasing appreciable quantities of radionuclides from the RCS through pre-existing SG tube leakage, if primary-to-secondary differential pressure is established. In order to prevent such a release, SG pressures must be maintained greater than RCS pressure when RCS activity is high.
C. Incorrect – this is the reason for reduced feed rate associated with a dry generator in the FR-H series of procedures. HFRH1BG rev 3 section 2.4
D. Incorrect – plausible as in E-3 step #7 check intact SG levels discuss isolating FW flow to prevent overfilling a SG.

Technical Reference(s):
1. E-2, Faulted Steam Generator Isolation, Rev 11
2. BD-E-2, Basis Document for E-2, Rev 4
3. HFRH1BG Rev 3, Section 2.4
4. BD-E-1, Basis Document for E-1, Rev 15
5. BD-E-3, Basis Document for E-3, Rev 14

References to be provided to applicants during examination: None

Learning Objective: T61.003D, emergency Operations, LP #15, Objective A & I:
A. EXPLAIN the Purpose and Major Action Categories of E -2, Faulted Steam Generator Isolation.

I. OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of E -2, Faulted Steam Generator Isolation.

Question Source: Bank # ___X L2636___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Station Blackout /SF 6	Group #	1		
	K/A #	00055 EK1.02		
	Importance Rating	4.1		
Knowledge of the operational implications of the following concepts as they apply to SBO: Natural circulation cooling				

Question # 12

Reactor Power was 100% when a loss of all AC power occurred. The crew is performing actions of ECA-0.0, Loss of All AC Power.

- TDAFP is running supplying 280,000 LBM/HR
- 'B', 'C', and 'D' SG NR levels are 2% and slowly rising
- Containment Pressure is 2 psig and slowly rising
- 'A' SG is ruptured and actions to isolate per ECA-0.0 Step #13 are complete

(1) To verify natural circulation is occurring, the reactor operator should verify the Loop 4 RCS COLD leg temperature is approximately ___(1)______.

And

(2) Per EOP Addendum 1, Natural Circulation indication also includes a RCS subcooling of at LEAST ......

A. (1) 561°F
(2) 30°F

B. (1) 561°F
(2) 50°F

C. (1) 565°F
(2) 30°F

D. (1) 565°F
(2) 50°F

Answer: A

Explanation:
Per EOP Addendum 1 Natural Circulation indication includes –
- *Core exit TCs - STABLE OR LOWERING*

- *RCS subcooling - GREATER THAN 30°F [50°F]*
- *RCS hot leg temperatures - STABLE OR LOWERING*
- *SG pressures - STABLE OR LOWERING*
- *RCS cold leg temperatures - CONSTANT AT OR SLIGHTLY ABOVE THE SATURATION TEMPERATURE FOR SG PRESSURE BEING MAINTAINED*

"SATURATION TEMPERATURE FOR SG PRESSURE BEING MAINTAINED". With no condenser available (therefore the steam dumps to the condenser are not available). and no ruptured SGs, ASDs should be controlling RCS temperature at the saturation temperature of the original setpoint of 1125 psig, 561F. If there was a ruptured SG, ASD setpoint would be adjusted to 1160 psig per ECA-0.0 step #13 RNO. The saturation temperature at this pressure is 565F.

For LOOP 2 and 3, the steam supply to the TDAFP is from the B and G SGs and these SG will be at some pressure less than their ASD setpoint.

No adverse containment exists (3.5 psig is not present) but it is above the EAM setpoint of 1.5 psig which makes the 50°F plausible but wrong.

A. Correct
B. Incorrect – See above explanation, wrong subcooling value
C. Incorrect – See above explanation wrong Cold Leg value
D. Incorrect - See above explanation, both are wrong

Technical Reference(s):
1. ECA-0.0, Loss of all AC Power, Rev 27
2. EOP Addendum 1, Natural Circulation Verification, Rev 2
3. ASME Steam Tables, Volume 83.

References to be provided to applicants during examination: ASME Steam Tables Compact Edition, Rev 83

Learning Objective: T61.003D, Emergency Operations, SD-03 (Simulator LP), Enabling Objective E: (URO-AEO-01) Perform a Natural Circulation Cooldown.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of Vital AC Instrument Bus	Group #	1		
	K/A #	00057 AA2.19		
	Importance Rating	4.0		
Ability to determine and interpret the following as they apply to Loss of Vital AC Instrument Bus and the following: The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus				

Question # 13

Reactor Power is 100% when the following events occur:

- Charging pump suction swaps from the VCT to the RWST
- Annunciator 82B, OTΔT Rod Stop, is LIT
- Annunciator 82C, OPΔT Rod Stop, is LIT
- Annunciator 83C, RX Partial Trip, is LIT

What event is in progress?

A. Loss of NK01

B. Loss of NN01

C. PR NI 41 is offscale HIGH

D. Loop 1 RCS RTD Channel Failure

Answer: B

Explanation:

Per OTO-NN-00001, Attachment C, the listed automatic actions are indicative of a loss of NN01. Loss of NK (125 VDC) is plausible as OTO-NK-00002 for a loss of NK01 references PZR level 459 and ensures that that instrument is not selected. Furthermore the NK bus is the normal power supply to the NN bus via an inverter which makes the loss of NK bus plausible also.

Per OTO-BB-00004, annunciator 82B, 82C and 83C are symptoms of a RCS RTD channel failure but there would be no swapover to the RWST making it plausible but incorrect.

A PR NI failing high would cause the annunciators also and it may be believed that it will cause a swapover to the RWST. The SR channel, SR N31, powered from NN01 will cause a swapover to the RWST if it fails high and isnt blocked

NOTE: Loss of NN04 would also be a correct answer due to VCT channel 185 and PR NI 44.

A. Incorrect – See above explanation
B. Correct
C. Incorrect – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):
1. OTO-NN-00001, Loss of Safety Related Instrument AC Power, Rev 38
2. OTO-BB-00004, RCS RTD Channel Failures, Rev 21

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #27, Objective B: DESCRIBE symptoms or entry conditions for OTO-NN-00001, Loss Of Safety Related Instrument Power.

Question Source: Bank # ______
Modified Bank # _X R11860_____
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 1
	Tier #	1		
Loss of DC Power	**Group #**	1		
	K/A #	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.				

Question # 14

Reactor Power is 100%.

- The main feeder breaker to Load Center NG01 trips on overcurrent
- NO operator action has been taken

What is the effect on the unit electrical alignment?

A. Battery Charger NK25 is powered and 120 VAC Bus NN01 is supplied by Battery NK11

B. Battery Charger NK21 has lost power and 120 VAC Bus NN01 is supplied by Battery NK11

C. Battery Charger NK21 is powered and 120 VAC Bus NN01 has transferred to the alternate supply

D. Battery Charger NK25 has lost power and 120 VAC Bus NN01 has transferred to the alternate supply

Answer: B

Explanation:

A. Incorrect. NK25 is powered from NG01.
B. Correct. NK21 is powered from NG01 and has lost power with battery NK11 now supplying power to NN01 through NK01.
C. Incorrect. NK21 has lost power as it is powered by NG01.
D. Incorrect. NN01 has not transferred to alternate power. It is still being supplied by NK01. NK01 is what has lost its normal power supply and is now powered from battery NK11.

Technical Reference(s):
1. OTN-NK-00001 ADD 01, 125VDC Bus NK01 And Distribution System, Rev 005

References to be provided to applicants during examination: None

Learning Objective: T61.0110, LP-6, Obj A, Draw and Explain a one line diagram of the Safeguards Power System to include the components and subsystems listed in objective B with normal breaker alignments shown and Obj B, Describe the purpose and operation of the following Safeguards Power system components and subsystems:
5. 125 VDC System (NK)

Question Source: Bank # __16621__
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2013 Q#14__

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 1
	Tier #	1		
Loss of Nuclear Service Water	Group #	1		
	K/A #	062 AK3.02		
	Importance Rating	3.6		
Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS.				

Question # 15

Following a Safety Injection Signal (SIS), EFHV-39/41, ESW TRN A TO SERVICE WTR SYS, _________.

A. open to prevent UHS Pond overflow

B. close to prevent low suction pressure to AFW pumps

C. open to ensure proper ESW pressure to containment coolers

D. close to ensure proper UHS Pond inventory

Answer: D

Explanation:
ESW return valves to Service Water, EFHV 39 (41) (A train) and EFHV 40 (42) (B train), are normally open to return flow from the ESW components to the Service Water System. The two valves per train are in series. One of the two series valves has a Red Train power supply; the other valve has a Yellow Train power supply.
They receive a close signal on:

- *Safety Injection Signal or*
- *Shutdown Sequencer actuation (at Step 0).*

Note that on a LSP w/AFAS signal these ESW to Service Water return valves remain open and the return valves to the UHS stay closed. This will result in pumping UHS water to the Circ & Service Water Cooling Tower Basin. Operator action is required to prevent main cooling tower basin overflow
.
A. Incorrect. EFHV-39/41 close on SI signal
B. Incorrect. ‘EFHV-39/41 stay open on LSP with AFAS
C. Incorrect. EFHV-39/41 close on SI signal.
D. Correct. EFHV-39/41 close on SI signal to ensure ESW goes thru safety rated equipment and back to UHS Pond

Technical Reference(s):
1. M-22EF01, P&ID ESW System, Rev 81

References to be provided to applicants during examination: None

Learning Objective:T61.0110, Systems, LP-5, Obj. D,.DESCRIBE the operation of the Essential Service Water System under the following conditions:

1. Standby
2. Safety Injection Signal
3. Loss of Offsite Power
4. Low Suction Pressure to the Auxiliary Feedwater Pumps
5. Opposite train NB bus undervoltage with low flow to the CTMT coolers

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Generator Voltage and Electric Grid Disturbances	Group #	1		
	K/A #	077 AK2.02		
	Importance Rating	3.1		
Knowledge of the interrelations between generator voltage and electric Grid Disturbances and the following: Breakers, relays				

Question # 16

A Switchyard Grid disturbance has caused a degraded NB01 bus voltage of 3750 VAC and relay 152 to actuate. What is the FIRST plant response to this relay actuation?

A. Starts the emergency diesel

B. Gives a safety-related load shed

C. Actuation of Shutdown Sequencer

D. Opens the normal and alternate ESF bus supply breakers

Answer: D

Explanation:

A. Incorrect. EDG will start after NB01 is de-energized and from Under-voltage Relay 127
B. Incorrect. Under-voltage Relay 127 causes this to happen
C. Incorrect. This occurs after EDG is up to speed and voltage
D. Correct. Relay 152 Degraded Voltage Relay opens supply breakers t

Technical Reference(s):
1. OTA-RK-00016, Addendum 19E, Rev 003
2. E-22NF01, Load Shedding & Emergency Load Sequencing Logic, Rev 008

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #6, Objective C, and F:

C. IDENTIFY the Safeguards Power System Main Control Board (MCB) controls and indications and DESCRIBE how each is used to predict, monitor or control changes in the Safeguards Power System.

F EXPLAIN safeguards equipment load shedding and load sequencing including the components and time intervals during the following conditions:
2. Undervoltage (UV) signal

Question Source: Bank # ______
Modified Bank # _5253_
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR55.41(b)(5)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
LOCA Outside Containment	Group #	1		
	K/A #	W/E04 EK2.2		
	Importance Rating	3.8		
Knowledge of the interrelations (between the LOCA outside containment) and the following: Facility*s heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility				

Question # 17

A LOCA outside containment has resulted in a RCS subcooling of 0°F.

Attempts are being made to determine if the leak has been isolated in accordance with ECA-1.2, LOCA Outside Containment.

What is the PRIMARY indication that the completed actions have been successful?

A. ECCS flow lowering

B. RCS pressure rising

C. Pressurizer level rising

D. RVLIS indication rising

Answer: B

Explanation:
Step #3 of ECA 1.2, has the operator check if break is isolated by checking if RCS pressure is rising otherwise the operator is directed to ECA-1.1 in the RNO. The distractors are plausible based on other types of LOCAs e.g. SBLOCA for distractor of pressurizer level, LBLOCA for RVLIS indication with a voided head. ECCS flow rate lowering is plausible due to the fundamentals of centrifugal pumps and RCS pressure higher due to the leak being isolated.

A. Incorrect – See above Explanation
B. Correct
C. Incorrect – See above Explanation
D. Incorrect – See above Explanation

Technical Reference(s):
1. ECA-1.2, LOCA Outside Containment, Rev 7

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References to be provided to applicants during examination: None

Learning Objective:T61.003D, Emergency Operations, LP #14, Objective E & F:

E. STATE and EXPLAIN the parameters which are evaluated, including their Criteria and Basis, to transition from ECA-1.2 to other procedures.

F. OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of ECA-1.2.

Question Source: Bank # __X L7052____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2009_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Changed one bank question distractor of Aux Building Sump Lowering to RVLIS

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of Emergency Coolant Recirculation /	**Group #**	1		
	K/A #	W/E 11 G2.1.7		
	Importance Rating	4.4		
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.				

Question # 18

The crew is performing ECA-1.1, Loss of Emergency Coolant Recirculation, with the current plant conditions:

- RWST level is 40%
- CTMT Recirc Sump Level is 10 inches
- CTMT Pressure is 30 PSIG

Per ECA -1.1 step #7, what is the required combination for operation of the CTMT COOLERS and/or CTMT SPRAY PUMPS?

A. 0 or 1 CTMT Coolers in Slow speed with 2 CTMT Spray Pumps in service

B. 4 CTMT Coolers in Slow speed with 1 CTMT Spray Pump in service

C. Less Than 3 CTMT Coolers in Slow speed with 1 CTMT Spray Pump in service

D. 3 or 4 CTMT Coolers in Slow speed with NO CTMT Spray Pumps in service

Answer: B

Explanation: *Per step #7 of ECA-1.1, 0 or 1 CTMT cooler in service with both CTMT Spray Pumps in service is the correct answer.*

A. Incorrect – this is a combination of the other acceptable combination when the plant has RWST level greater than 36% and CTMT Pressure between 27 and 48 psig therefore making it plausible but wrong
B. Correct per the reference
C. Incorrect – this is an acceptable combination with the given CTMT pressure but when RWST level is between 6 and 36%
D. Incorrect – this is an acceptable combination with the given CTMT pressure but when RWST level is between 6 and 36%

Technical Reference(s):

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1. ECA-1.1, Loss of Emergency Recirculation, Rev 13

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #13, Objective H: OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of ECA-1.1, Loss of Emergency Coolant Recirculation.

Question Source: Bank # ______
Modified Bank # ______
New ____X___

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(8)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Inadequate Heat Transfer—Loss of Secondary Heat Sink / SF4	**Group #**	1		
	K/A #	W/E05 EK3.1		
	Importance Rating	3.4		
Knowledge of the reasons for the following responses as they apply to Loss of Secondary heat Sink: Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics				

Question # 19

A Reactor trip and Safety Injection have occurred.

- The crew has transitioned to FR-H.1, Response to Loss of Secondary Heat Sink
- ALL SG wide range levels are 35% and slowly trending down
- Total AFW flow is 0 lbm/hr
- Containment Pressure is 2.0 psig slowly trending up
- The crew is currently performing Step #7 and the Main Feed Pumps will NOT reset. The Startup Feed Pump is unavailable

Based on the above conditions and per FR-H.1, what is the NEXT action required and reason for the action?

A. Stop all RCPs to minimize the heat input into the RCS

B. Establish RCS Bleed and Feed due to the loss of secondary heat sink

C. Depressurize one SG to less than 550 psig to allow feed with condensate

D. Return to Step 1 of FR-H.1 and continue to try to restore feed to any generator before any generator is dry

Answer: C

Explanation:

A. Incorrect – Stopping all RCPs is an action in the continuous action step #2 and is the entire Step #6 which makes it plausible. Per the stem the crew is at step #7 and RCPs would have already been stopped. If the specific order of the step is not remembered and/or it is believed that a loss of secondary heat sink has occurred; RCS Bleed and Feed is required stopping RCP

to minimize heat input is plausible. It is wrong as SG level is greater than the 27% (given SG level is 35%) but the adverse value of 42% may be remembered (adverse containment of 3.5 has not occurred), If SG was less than 27% [42%], Step #2 would require stopping the RCPs and initiation actions for Bleed and Feed.
B. Incorrect – see explanation for #1 on why RCS Bleed and Feed is not required at this time
C. Correct – this is the NEXT action per FR-H.1 specifically step #9
D. Incorrect – this is the RNO of step #11 which directs the operator back to the beginning steps of FR-H.1 to continue action to restore feed to any generator. This is also wrong as the SG level is 27% but the Dry definition is 10% [25%].

Technical Reference(s):
1. FR-H.1, Response to Loss of Secondary Heat sink, Rev 18

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #26, Objective I OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of:
1. FR-H.1, Response To Loss Of Secondary Heat Sink.

Question Source: Bank # _____
Modified Bank # _____X_L16518______
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Loss of Intermediate Range Nuclear Instrumentation	Group #	2		
	K/A #	033 G2.1.28		
	Importance Rating	4.1		
Knowledge of the purpose and function of major system components and controls				

Question # 20

Reactor Power is 8%.

Intermediate range nuclear instrument channel N-35 fails full scale LOW.

As a result of the failure, the associated source range channel (N31) will ___(1)___.

Per OTO-SE-00001, Nuclear Instrument Malfunction, the crew will ___(2)___.

A. (1) energize
(2) raise reactor power to greater than P-10 within 24 hours

B. (1) energize
(2) suspend operations involving positive reactivity additions

C. (1) remain de-energized
(2) raise reactor power to greater than P-10 within 24 hours

D. (1) remain de-energized
(2) suspend operations involving positive reactivity additions

Answer: C

Explanation:
Per OTO-SE-00001, Attachment B, Step B2 RNO, as power was less than P-10 and ONLY one IR was inoperable, one of the options is to raise power to greater than 10% within 24 hours. The Distractor of suspend operations involving positive reactivity additions is if power is less than P-10 (which is in the stem) and if BOTH IR channels are inoperable.
Permissive P-6 allows manual blocking of the SR when 1 of 2 IR channels is above 10^{-10} amps. This ensures that the operator has IR indication available before de-energizing the SR by observing one decade (10^{-11} 1o 10^{-10} amps) of IR overlap.P-10 auto blocks the SR HV.
At the power given in the stem and per normal operating procedures, the SR will have been manually blocked and would not reenergize as the other NI channel is greater than 10^{-10} amps. Energize is plausible if it is believed, the coincidence is any IR less than 10^{-10}

A Incorrect – See above explanation – energize is incorrect
B. Incorrect – See above explanation – both are wrong
C. Correct
D. Incorrect – See above explanation – procedure action is incorrect

Technical Reference(s):
1. OTO-SE-00001, Nuclear Instrument Malfunction

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #42, Objective D: Given a set of plant conditions or parameters indicating a Nuclear Instrument Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # _7162__
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	2		
Steam Generator (S/G) Tube Leak /	**Group #**	1		
	K/A #	00037 AK1.01		
	Importance Rating	2.9		
Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Use of steam tables.				

Question # 21

The crew is responding to an 'A' Steam Generator Tube Leak using OTO-BB-00001, 'Steam Generator Tube Leak', with the following plant conditions:

- 'A' S/G pressure is 965 psig
- 'B' S/G pressure is 925 psig
- 'C' S/G pressure is 925 psig
- 'D' S/G pressures is 925 psig
- RCS depressurization is in progress

What is the HIGHEST indicated core exit temperature that provides 30°F subcooling after completion of the RCS depressurization?

A. 507°F

B. 512°F

C. 535°F

D. 542°F

Answer: B

Explanation: See below

A: Incorrect: The wrong steam generator was chosen, using 940 psia. This would yield a 537°F Sat. Temp, and then with 30°F subcooling, it would be 507°F.
B: Correct: The correct steam generator was chosen, using 980 psia. This would yield a 542°F Sat. Temp, and then with 30°F subcooling, it would be 512°F.
C: Incorrect: The wrong steam generator was chosen, using 925 psig vice 940 psia. This would yield a 535°F Sat. Temp, but does not take in to account the 30°F subcooling.
D: Incorrect: The correct steam generator was chosen, using 980 psia. This would yield a 542°F Sat. Temp, but does not take in to account the 30°F subcooling.

Technical Reference(s):
1. OTO-BB-00001, SG Tube Leak, Rev 23
2. ASME Steam Tables Volume 83, 2006 edition

References to be provided to applicants during examination: ASME Steam Tables Volume 83, 2006 edition

Learning Objective: GFES Thermodynamics, Chapter 3, Steam Objective 8 Apply saturated and superheated steam tables in solving liquid-vapor problems.

Question Source: Bank # __X_ ADAMS database_Point Beach_ ML13028A065-023__and 2016 Audit Q#22_____
Modified Bank # ______
New ______

Question History: Last NRC Exam ___N/A for Callaway - _Point Beach 2012________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(8)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	1		
Area Radiation Monitoring System Alarms	Group	2		
	K/A	061 AK2.01		
	Importance Rating	2.5		
Knowledge of the interrelations between the Area radiation Monitoring System Alarms and the following: Detectors at each ARM system location				

Question # 22

The N-16 Main Steamline monitors have an RM-80 Microprocessor Unit associated with them that is located in the vicinity of its associated detector.

What is a function of the N-16 RM-80 Microprocessor?

A. Allows the detector to be source checked

B. Indicate local radiation levels from the detector

C. Transmits its data to RM-23 and/or RM-11

D. Automatically actuate plant equipment on an ALERT or HIGH alarm condition

Answer: C

Explanation:
Each monitor has one RM80 assembly (except as noted of the N-16 monitors) which receives pulse signals from the detector assembly. The RM80 is a microprocessor which processes the signals in digital form, computes averages for 10 minutes, 1 hour, 1 day, and 28 days and stores this data. This unit is normally mounted on the same skid as that of a detector assembly, but in some cases it is remotely located.
The RM80 will transmit its data to either a RM23, the RM11, or in most cases both, upon request. It also can produce analog signals for recorders and has built in diagnostics to assure rapid and reliable failure detection and isolation.. In addition, RM80s are equipped with a local audible alarm and alarm acknowledge push-button, however, other than pump and valve controls and a sample flow indicator, there is no capability for operator interface. Nor is there any local indicator of radiation level. Operator interface as well as indication is provided by the RM23 and/or the RM11.

A: INCORRECT: N-16 RM-80s are shared between 2 detectors and does not allow the detectors to be source checked
B: INCORRECT: N-16 RM-80s are shared between 2 detectors and does not indicate local radiation levels from the detectors
C: CORRECT:, RM-80s transmits data to RM-23 and/or RM-11
D: INCORRECT There are no automatic actions associated with this Rad Monitor:

Technical Reference(s):
1. OTN-SP-00003 Remote Control and Display Module Operations, Rev 5
2. OTA-SP-RM011, Radiation Monitor Control Panel RM-11, Rev 43

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #36, Objective A: DESCRIBE the purpose and operation of the following Process and Area Radiation Monitoring components:
8. RM80 Microprocessor

Question Source: Bank # ________
Modified Bank # ____
New __X___

Question History: Last NRC Exam ___N/A___

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier	1		
Control Room Evacuation	**Group**	2		
	K/A	00068 AA2.03		
	Importance Rating	4.0		
Ability to determine and interpret the following as they apply to control room evacuation: T-hot, T-cold, and in-core temperatures				

Question # 23

The Reactor has been tripped from 100% power, and the Control Room has been evacuated due to smoke.

Steps 1-10 of OTO-ZZ-00001, Control Room Inaccessabilty, have been successfully completed, and personnel are arriving at their remote stations (no local actions have been taken).

What correctly describes expected current plant conditions?

A. Forced circulation is in operation. RCS T-cold temperatures should be trending at Tsat for SG pressures.

B. Forced circulation is in operation. RCS T-hot temperatures should be trending at Tsat for SG pressures.

C. Natural circulation should be in progress. RCS T-cold temperatures should be trending at Tsat for SG pressures.

D. Natural circulation should be in progress. RCS T-hot temperatures should be trending at Tsat for SG pressures.

Answer: C

Explanation:
A: INCORRECT - Plausible because forced circulation would be in progress if the RCPs were not tripped.
B: INCORRECT - Plausible because forced circulation would be in progress if the RCPs were not tripped. T-hot/SG press comparison is plausible if the applicant does not understand the RCS/SG relationship (Thot must be higher than SG pressure for thermal driving head
C: CORRECT - RCPs are tripped prior to leaving the Control Room and with natural circ established, SG pressure trend with Tcold.
D: INCORRECT - Plausible if the applicant does not understand the RCS/SG relationship (Thot must be higher than SG pressure for thermal driving head)

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Technical Reference(s):
1. OTO-ZZ-00001, Control Room Inaccessibilty, Rev 47

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #31, Objective E. Given a set of plant conditions or parameters indicating Control Room Inaccessibility, IDENTIFY the correct procedure(s) to be utilized and OUTLINE the high level actions to stabilize the plant.

Question Source: Bank # _
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Inadequate Core Cooling	Group #	2		
	K/A #	074 EA1.01		
	Importance Rating	4.2		
Ability to operate and monitor the following as they apply to Inadequate Core Cooling: RCS Water Inventory				

Question # 24

The crew has entered FR-C.1, Response to Inadequate Core Cooling, and is performing step #1, Check ECCS Valves – Proper Emergency Alignment.

Current plant conditions are as follows:

- RCPs are OFF
- RVLIS Pumps OFF indication – 30% slowly rising
- ECCS flow is 0 lbm/hr
- Core Exits TCs - 800°F slowly rising
- Adverse CTMT conditions exist
- All NR SG levels are 0%
- Total AFW flow is 300,000 lbm/hr

Per FR-C.1, what is the NEXT action the crew should take?

A. Start a RCP

B. Depressurize All Intact SGs to 220 psig

C. Try to establish ECCS flow

D. Ensure Hydrogen Analyzers in service

Answer: C

Explanation:

A. Incorrect but plausible as step #3 checks for RCP support systems available and an RCP is started in step #18. Step #18 is reached from the RNO on step #5 and then at step #9 the RNO directs go to step #18 to establish a heat removal method.
B. Incorrect but plausible as this action is step #11 and is reached from the RNO in step #5 with the rest of the actions implemented successfully.
C. Correct – per step #2 RNO, as ECCS flow is not established, the direction is to try to establish ECCS flow using the normal charging line. Furthermore at step #6 the RNO will be implemented

which directs return to step #1 and therefore step #2 RNO will be utilized as RVLIS level is less than 42%.
D. Incorrect but plausible as this is done in step 8 or its RNO

Technical Reference(s):
1. FR-C.1, Response to Inadequate Core Cooling, Rev 10

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #25, Objective I: OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of:
1. FR-C.1, Response To Inadequate Core Cooling.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
W E01 & E02) SI Termination / 3	Group #	2		
	K/A #	W/E02 EK3.2		
	Importance Rating	3.0		
Knowledge of the reasons for the following responses as they apply to SI termination: Normal, abnormal and emergency operating procedures associated with (SI Termination)				

Question # 25

Reactor Power was 100% when a Small Break LOCA occurred.

The crew is now performing E-1, Loss of Reactor or Secondary Coolant, and are performing step 6, CHECK If ECCS Flow Should Be Reduced.

Current plant conditions are as follows:

- Steam Generators NR levels are 21%, 22%, 24% and 28%
- Total AFW flow is 170,000 lbm/hr
- RCS Pressure is 1200 psig and stable
- Containment Pressure is 5 psig and stable
- PZR level is 40% and stable
- RCS subcooling is 55°F and stable

(1) Based on the above conditions, the crew should implement/continue to ___(1)___ procedure FIRST.

And

(2) Why?

A. (1) E-1, Loss of Reactor or Secondary Coolant
(2) Plant not stabilized to exit from current procedure

B. (1) ES-1.1, SI Termination
(2) All conditions met to reduce ECCS flow

C. (1) ES-1.2, Post Loca Cooldown And Depressurization
(2) All conditions met to cooldown RCS

D. (1) FR-H.1, Response To Loss Of Secondary Heat Sink
(2) Secondary heat sink is inadequate

Answer: B

Explanation:
E-1, Stpe 6, CHECK If ECCS Flow Should Be Reduced:
- *RCS subcooling – GREATER THAN 30°F [50°F]*
- *Secondary heat sink:*
- *Narrow range level in at least one intact SG GREATER THAN 7% [25%]*

OR
- *Total feed flow tointact SGs – GREATER THAN 270,000 LBM/HR*
- *RCS pressure - STABLE OR RISING*
- *PZR level - GREATER THAN 9% [29%]*

A. Incorrect – See above explanation
B. Correct – See above explanation
C. Incorrect – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):
1. ES-1.1 SI Termination, Rev 16
2. BD-ES-1.1, Basis Document for ES-1.1, Rev 4

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, Objective H & I:

H. STATE and EXPLAIN the parameters which are evaluated, including their Criteria and Basis, to transition from ES-1.1 to other procedures.

I.OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of ES-1.1.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
Steam Generator Over-pressure / 4	Group #	2		
	K/A #	W/E 13 EA1.3		
	Importance Rating	3.1		
Ability to operate and/or monitor the following as they apply to Steam Generator Overpressure: Desired operating results during abnormal and emergency conditions				

Question # 26

The crew entered FR-H.2, Steam Generator Overpressure, due to an overpressure condition on 'B' S/G.

- Operators have just completed Step #2 in which Main Feedwater Isolation to the 'B' S/G has been verified
- 'B' S/G narrow range level is 80% and slowly rising
- RCS Hot legs Temperatures are 550°F and steady
- 'A', 'C', and 'D' S/G NR levels are 25% and slowly rising
- AFW flow to each S/G is as follows:
  - 'A' S/G 80,000 lbm/hr
  - 'B' S/G 20,000 lbm/hr
  - 'C' S/G 80,000 lbm/hr
  - 'D' S/G 90,000 lbm/hr

(1) Based on the above conditions and in accordance with FR-H.2, the NEXT action the crew should take is to dump steam from the affected S/G …..

And

(2) Per FR-H.2 step #5 and in order to 'return to the procedure and step in effect', the crew should control the affected S/G pressure less than a MAXIMUM of ...

A. (1) by establishing SG blowdown
(2) 1160 psig

B. (1) by establishing SG blowdown
(2) 1234 psig

C. (1) using MSIV Bypass Valves and Condenser steam dumps
(2) 1160 psig

D. (1) using MSIV Bypass Valves and Condenser steam dumps
(2) 1234 psig

Answer: D

Explanation:
Establishing S/G blowdown is additional method to dump steam in step #8 of FR-H.2. This option is not available in step #4 as a method to dump steam from the affected SG but is an option to remove pressure later in the procedure and in physical layout of the plant which therefore makes it a plausible choice. Using the MSIV Bypass Valves and Condenser steam dumps is the NEXT action per step #4 of FR-H.2. Per step #5c of FR-H.2, the crew will " CONTROL steam release to maintain SG pressure less than 1234 PSIG". The distractor of 1160 psig is a distractor from E-3 Tube rupture step #3 and is plausible as it is a S/G pressure setting important in the performance of an EOP.

A. Incorrect – See explanation above
B. Incorrect – See explanation above
C. Incorrect – See explanation above
D. Correct – See explanation above

Technical Reference(s):
1. FR-H.2, Response to Steam Generator Overpressure, Rev 6

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #26, Objective G & I:

G. STATE and EXPLAIN the parameters which are evaluated, including their Criteria and Basis, to transition from the following procedures to another procedure.
2. FR-H.2, Response To Steam Generator Overpressure.

I: OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of:
2. FR-H.2, Response To Steam Generator Overpressure.

Question Source: Bank # ______
Modified Bank # __X no bank id yet – modified from 2017 audit Q#26_
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(10)

Comments:
k/a match as the question asks what is the desired method to dump steam from the affected S/G (ability to operate to achieve the desired results of lowering the affected S/G pressure). Additionally, the question also asks at what affected S/G pressure can the FR procedure be exited (which is the desired operating result i.e. lowered the pressure such that the condition no longer exists).

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	1		
W/E 10 Natural Circulation/4	**Group #**	2		
	K/A #	W/E10 EK3.1		
	Importance Rating	3.4		
Knowledge of the reasons for the following responses as they apply to natural Circulation with Steam Void in Vessel with/without RVLIS: Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics				

Question # 27

A Natural Circulation RCS cooldown is in progress per ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS), due to a loss of offsite power. No RCPs are running.

Why is Pressurizer Level controlled to LESS THAN 90% in step #5 of ES-0.3?

A. To prevent uncovery of Pressurizer Heaters

B. To prevent a water solid Pressurizer

C. To allow conditions to be established to try to start an RCP

D. To allow for excess letdown to be established

Answer: B

Explanation:

A. Incorrect – plausible as PZR heater turn off and letdown isolate at 17% PZR level. This is basis for step #5 level of greater than 25%
B. Correct – This is facilitated by step #5 which directs a band of 25% to 90% for the reasons. 90% is to prevent a water solid PZR and the resultant loss of pressure control..
C. Incorrect – plausible as per Step #1 RNO if RVLIS indication is less than 100% (pumps off) then PZR level is raised to 90%. Therefore the candidate may assume that this PZR level band required to start a RCP if RCS level is low.
D. Incorrect – Step #3 is "Establish PZR level to accommodate Void Growth".RNO controls Charging and Letdown as necessary.

Technical Reference(s):
1. ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS), Rev 15
2. EOP Addendum 3, Starting an RCP, Rev 3
3. BD-ES-0.3 Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS) Rev 004

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References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #7, ES-0.2, ES-0.3, ES-0.4 Natural Circulation Objective C &H:

C. DESCRIBE the requirements and basis for the Continuous Action Steps of:
1. ES-0.2.
2. ES-0.3.
3. ES-0.4.

H: OUTLINE procedural flow path including major system and equipment operation in accomplishing the goal of the following procedures:
1. ES-0.2
2. ES-0.3
3. ES-0.4

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	1		
Emergency Boration	Group	2		
	K/A	024 G2.1.7		
	Importance Rating	4.4		
Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation				

Question # 28

Reactor Power was at 100% when the turbine generator spuriously tripped.

- Both reactor trip breakers are CLOSED
- The crew is performing the actions of FR-S.1, Response to Nuclear Power Generation/ATWS
- Emergency boration was begun 20 minutes ago in accordance with FR-S.1 requirements
- Tave is at 547°F and slowly lowering

Per FR-S.1, the crew can stop emergency boration if ___________?

A. RCS boron is 2000 ppm

B. the reactor is tripped

C. RCS temperature is below 557°F

D. all dilution paths are isolated

Answer: A

Explanation:

A CORRECT: FR S.1 Caution before step 16 states "Boration should continue to obtain adequate shutdown margin during subsequent actions Max SDM per Table 1-8 of curve book is 1691 ppm
B Incorrect: There is no procedural direction to stop boration when the reactor is tripped but plausible as enough negative reactivity from the rods has been added to shutdown the reactor.
C Incorrect: The RCS is below normal Tave, but that is the reason for borating in ES-0.1, not in FR S.1. There is procedural guidance to deal with an uncontrolled cooldown.
D Incorrect: There is no procedural direction to stop boration when dilution paths are isolated but plausible since positive reactivity would no longer be added
".

Technical Reference(s):

1. FR-S.1, Response to Nuclear Power Generation/ATWS, Rev 12

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #29 Objective K. OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of:
1. FR-S.1, Response to Nuclear Power Generation/ATWS.

Question Source: Bank # _
Modified Bank # _____
New ___X___

Question History: Last NRC Exam ____N/A____

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Reactor Coolant Pump	Group #	1		
	K/A #	003 A1.01		
	Importance Rating	2.9		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCP controls including: RCP vibration				

Question # 29

Reactor Power is 30%.

- RCP "C" frame vibration is 4 MILS and rising at 1.5 Mil/hr
- RCP "C" shaft vibration is 14 MILS and rising at 1.5 Mil/hr
- RCP "C" No. 1 Seal Leakoff Flow is 4 gpm and rising at 2 gpm every hour
- RCP "C" No. 1 Seal and Bearing Inlet Temperature is 200°F

What parameter requires an immediate shutdown/trip of RCP "C"?

A. shaft vibration

B. frame vibration

C. No. 1 Seal Leakoff Flow

D. No. 1 Seal And Bearing Inlet Temperature

Answer: B

Explanation:

Per OTO-BB-00002, step #B2 "CHECK No. 1 Seal Leakoff flow On All RCPs - LESS THAN 6 GPM ", and with the data given in the stem (4 gpm and rising at 2 gpm every hour), it is plausible that this would require a RCP shutdown but wrong as this would be in one hour from now, not an immediate shutdown.

Per OTO-BB-00002, step #A1 "CHECK RCP Vibration Level"
- *ALL RCPs vibration on the frame - LESS THAN 5 MILS*
- *ALL RCPs vibration on the shaft - LESS THAN 20 MILS*

These are both met and proceeding to step #A2
- *ALL RCPs vibration on the frame - LESS THAN 3 MILS*
- *ALL RCPs vibration on theshaft - LESS THAN 15 MILS*

As the Frame is in excess of 3 MILS and it rate is higher than 1 MIL per hr but RX power is less than 48% the RCP is tripped per the RNO directing Attachment E.

Per OTO-BB-00002, step #B4, "CHECK No. 1 Seal & Bearing Inlet Temperature - LESS THAN 230°F ON ALL RCPs" the RNO of this step would direct securing the RCP but as temperature is @200F this doesn't apply. Per step #B5, CHECK No. 1 Seal & Bearing Inlet Temperature - LESS THAN 200°F ON ALL RCP" the RNO is applicable which requires monitor and contacting engineering. This step is the source of the 200F in the stem.

A. Incorrect – See above explanation
B. Correct
C. Incorrect – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):
1. OTO-BB-00002, RCP Off Normal, Rev 33

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off normal Operations, LP #11, Objective D: Given a set of plant conditions or parameters indicating a RCP Off-Normal condition, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Reactor Coolant Pump	Group #	1		
	K/A #	003 G.2.4.6		
	Importance Rating	3.7		
Knowledge of EOP mitigation strategies				

Question # 30

What is the reason for tripping the Reactor Coolant Pumps during a Small Break Loss of Coolant Accident (SBLOCA) when the appropriate plant parameters are met?

A. To minimize additional heat input into the RCS

B. To prevent excessive depletion of RCS water inventory

C. To prevent emergency boration flow from becoming diluted in RCS loops

D. To minimize the challenge to the Pressurizer PORVs and associated block valves

Answer: B

Explanation:
Per the ERG Executive Volume reference page 11, the correct answer is that "The reason for purposely tripping the RCPs during accident conditions is to prevent excessive depletion of RCS water inventory through a small break in the RCS which might lead to severe core uncovery if the RCPs were tripped for some reason later in the accident. The RCPs should be tripped before RCS liquid inventory is depleted to the point where tripping of the pumps would cause the break to immediately uncover."

The distractors of to minimize the additional heat input into the RCS is the longer term recovery from many accidents and not correct as the stem is focused on SBLOCA reasons (page 3 of reference)

A. Incorrect – but plausible as this a longer term recovery from many accidents and not correct as the stem is focused on SBLOCA reasons
B. Correct – See above explanations
C. Incorrect – Plausible if the candidate believes that the SI injection flow and therefore boron would be diluted in the RCS loops' volume and this is a reason to trip the RCPs as a SI or CCP pump in service is a requirement to trip the RCPs. The SI or CCP pump suction will be from the RWST which is a highly borated volume therefore the RCPs may be pumping/moving this emergency boration flow into the RCS loops effectively diluting the RPV
D. Incorrect – plausible if the candidate believes tripping the RCPs will minimize PZR Pressure challenges and would prevent opening of the Pressurizer PORVs. Per Page 9 of the reference keeping the RCPs running (i.e. not tripping them) "is desirable to keep the RCPs running during

these transients to: 1) maintain normal pressure control using pressurizer spray and thereby avoid opening of the pressurizer PORVs". Plausible as this is an opposite reason as asked for in the stem and related as it is a reason why the RCP trip parameters setpoints are chosen to ensure they are reached during a SBLOCA but allow the RCPs to remain in operation during other transients

Technical Reference(s):
1. E-0, Reactor Trip or Safety Injection, Rev 20
2. BD-E-0, Basis Document of E-0, Rev 10
3. ERG Generic Issue RCP Trip / Restart Rev 3 March 31, 2014

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #4, E-0, Objective G: DESCRIBE the Criteria and the Basis for information as stated on the E-0, Reactor Trip Or Safety Injection, Foldout Page.

T61.003D, Emergency Operations, LP #2, Executive Volume, Objective A: DESCRIBE the bases for the RCP trip criteria.

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Chemical and Volume Control	Group #	1		
	K/A #	004 K6.37		
	Importance Rating	2.9		
Knowledge of the effect of a loss or malfunction on the following CVCS components: Boron loading of demineralizer resin				

Question # 31

Reactor Power is 100%.

BG TCV-130, Letdown Heat Exchanger CCW Temperature Control Valve, is slowly failing OPEN.

What is/are the effect(s) of this failure?

A. T_{avg} will slowly rise

B. T_{avg} will slowly lower

C. Channeling of the CVCS demineralizers

D. BG-TCV 129 will divert letdown flow away from the CVCS demineralizers

Answer: A

Explanation:
Per step #3.1.5 of Addendum 4 " Changes in CCW system temperature or RCS Letdown Flow can affect core reactivity by causing changes in letdown temperature. The letdown demineralizers release boron on rising temperatures and absorb boron on lowering temperatures. Therefore, when operations are in progress which alters CCW temperature or RCS Letdown Flow to the letdown heat exchanger, the operation of BG TK-130, LTDN HX OUTLET TEMP CTRL should be closely monitored such that temperature on BG TI-130, LTDN HX OUTLET TEMP is maintained stable, or slowly changed following letdown flow changes. Manual operation of BG TK-130 may be required."

Therefore, since more CCW flow is being admitted to the HX the CVCS outlet temperature will be lower which will result in positive reactive. To compensate for this positive reactivity, Tavg will have to rise to insert negative reactivity.

A. Correct – See above explanation
B. Incorrect – See above explanation
C. Incorrect but plausible as when letdown temperature changes the density and therefore flow changes. Furthermore OTN-BG-00001 Addendum 4 step 3.1.3 " Letdown flow should NOT

exceed 120 gpm. Controlling letdown flow at 120 gpm nominally prevents channeling flow through the CVCS demineralizer resin, reduces fatigue transients of ASME piping and components, and prevents excessive flow to the CVCS heat exchangers.
D. Incorrect but plausible as TCV 129 will divert on a letdown HX outlet temp of 137F which could happen if BG TCV 130 fails CLOSED not open.

Technical Reference(s):
1. OTN-BG-00001, Chemical and Volume control System , Rev 57
2. OTN-BG-00001, Addendum 4, Operation of CVCS Letdown, Rev 24

References to be provided to applicants during examination: None

Learning Objective: T61.0110, systems, LP #11, CVCS, Objective V: OTN-BG-00001, Chemical and Volume Control System
1. STATE the precautions and limitations and bases pertaining to the following:
   a. Starting a CCP or the NCP.
   b. Required action if normal letdown is lost with RCS > 350F.
   c. Maintaining seal injection flow.
   d. Placing the Cation bed in service.
   e. Maximum RCS temperature entering demins.
   f. Maximum letdown flow.
   g. CCW temperature or letdown flow change effects upon reactivity.
   h. Letdown Throttle valve bypass valve limitations.

Question Source: Bank # __ ____
Modified Bank # ____X__L14488______
New _______

Question History: Last NRC Exam _N/A for Callaway–version appeared on DG 2010 ILT Exam_

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Modified the bank question to focus on what a reactor operator would observe or be expected to understand about the reactivity impacts of letdown temperature.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Emergency Core Cooling	Group	1		
	K/A	006 K5.11		
	Importance Rating	2.5		
Knowledge of the operational implications of the following concepts as they apply to ECCS: Basic heat transfer equation				

Question # 32

A Large Break Loss of Coolant Accident has occurred and ECCS is operating in the Cold Leg Recirculation mode.

Based on standard heat transfer relationships, what describes the INITIAL effects of RAISED CCW temperature to the RHR Heat Exchanger during these plant conditions?

ΔT across the RHR Heat Exchanger tubes will…

A. RISE resulting in LOWER heat transfer rate in the RHR Heat Exchanger

B. LOWER resulting in LOWER heat transfer rate in the RHR Heat Exchanger

C. RISE resulting in HIGHER heat transfer rate in the RHR Heat Exchanger

D. LOWER resulting in HIGHER heat transfer rate in the RHR Heat Exchanger

Answer: B

Explanation:
A: INCORRECT: Higher tube inlet temperature results in ΔT across the HX going down.
B: CORRECT: Based on Q=UA ΔT, if ΔT goes down, heat transfer rate also goes down.
C: INCORRECT: Higher tube inlet temperature results in ΔT across the HX going down.
D: INCORRECT: Based on Q=UA ΔT, if ΔT goes down, heat transfer rate must also go down.

Technical Reference(s):
1. M-22EM01(Q), P&ID, HPCI, Rev 39
2. M-22EP01, P&ID Accumulator Safety Injection, Rev 18

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #56, Objective E &H:

E. EXPLAIN the purpose of Cold Leg Recirculation.

H. DRAW and/or LABEL the normal ECCS lineup and flowpath for Cold Leg Recirculation.

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Question Source: Bank # _ X_ not used at Callaway before _
Modified Bank # __ __
New _______

Question History: Last NRC Exam _STP NRC ILT Exam for 2017___

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(14)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Emergency Core Cooling	Group	1		
	K/A	006 G2.4.9		
	Importance Rating	3.8		
Knowledge of low power / shutdown implications in accident (e.g loss of coolant accident or loss of residual heat removal) mitigations strategies				

Question # 33

The Plant is in MODE 3 with RCS pressure stable at 800 psig.

- When RO reports that Pressurizer level is 5% and lowering

The crew will enter ____(1)_____ and the FIRST action is _____(2)______.

A. (1) OTO-BB-00010, Shutdown LOCA
(2) secure RCP's

B. (1) E-0, Rx Trip or Safety Injection
(2) secure RCP's

C. (1) OTO-BB-00010, Shutdown LOCA
(2) actuate Phase A

D. (1) E-0, Rx Trip or Safety Injection
(2) actuate Phase A

Answer: A

Explanation:

A CORRECT: OTO-BB-00010, Shutdown LOCA. Is entered because SI accumulators are isolated and first step is stopping RCPs
B: INCORRECT: E-0 Is not entered because SI accumulators are isolated.
C: INCORRECT: OTO-BB-00010, Shutdown LOCA. Is entered because SI accumulators are isolated and step 8 directs actuation of Phase A
D: INCORRECT: E-0 is not entered because SI accumulators are isolated.

Technical Reference(s):
1. OTO-BB-00010, Shutdown LOCA, Rev 10

References to be provided to applicants during examination: None

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Learning Objective: T61.003B, LP #64, Objective A: STATE the purpose and major action categories of OTO-BB-00010, Shutdown LOCA

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Pressurizer Relief/Quench Tank	Group #	1		
	K/A #	007 A3.01		
	Importance Rating	2.7		
Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT				

Question # 34

Reactor Power is 100%.

- PRT temperature is 120°F and rising slowly
- PRT level is 87% and rising slowly
- PRT pressure is 8 psig and rising slowly

Leakage through what valve could be causing these Pressurizer Relief Tank (PRT) conditions?

A. CVCS Letdown Relief, BG8117

B. ECCS Accumulator Reliefs, EP8855A-D

C. Safety Injection Pump Suction Relief, EM8858A

D. PRT Reactor Makeup Water Supply, BBHV8045

Answer: A

Explanation:

Per the reference, the following relieve to the PRT: RHR suction, RCP seal leakoffs, CVCS Letdown. Additionally, the Reactor Makeup Water also can flow to the PRT through BBHV8045.

A. Correct – per the reference and above explanation
B. Incorrect – plausible as if any of these reliefs were leaking by some or all of the above indications would be present making it plausible but these reliefs relieve directly to CTMT atmosphere per M-22EP01.
C. Incorrect – plausible as this would cause some of the above indications (Level, pressure) making it plausible however per M-22EM01 these reliefs discharge to the Recycle Hold Up Tank.
D. Incorrect – plausible because if this valve was leaking by, PRT level and pressure would be slowly rising but Temperature would be constant or slowly lowering as makeup water would be less than the 120F (given in the stem).

Technical Reference(s):
1. M-22BB02, P&ID Reactor Coolant System, Rev 33

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems LP #9, Objective E: IDENTIFY the inputs to the PRT.

Question Source: Bank # __X – no bank id ____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2017 ILT NRC Exam Q#52________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

There is no automatic operation of the PRT system (i.e its function as a quench tank) except for the relief disc rupture. All makeup and draining is done manually, Valve leak by is the only method for PRT mass addition without manual operation. k/a/ match as the candidate is given PRT parameters to monitor and based on these parameters, the candidate must determine the source i.e which component is discharging into the PRT.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Component Cooling Water	**Group #**	1		
	K/A #	008 K4.01		
	Importance Rating	3.1		
Knowledge of the design feature(s) and/or interlocks which provide for the following: Automatic start of standby pump				

Question # 35

With PEG01C, the 'C' CCW Pump, in service, what would cause the automatic start of PEG01A, the 'A' CCW Pump?

A. Safety Injection Signal (SIS)

B. Starting PBG05B, the 'B' CCP

C. PEG01C's, 'C' CCW Pump, discharge pressure lowers to 35 psig

D. Loss of the NB01 bus and subsequent loading of 'A' EDG

Answer: D

Explanation:
The following are the auto start signals available for the CCW pumps:

- Starting a CCP with no CCW Pump running in that train
- Low discharge pressure of 29 psig after 4 sec time delay, if the parallel pump (opposite train) was running.
- Activation of the Shutdown or LOCA Sequencers

Per E-22NF01, at time 5 seconds with the shutdown sequencer actuated, the A CCW pump will autostart and if it wouldn't start the standby (i.e C CCW pump would start at the 10 second point.

A. Incorrect –signal to start the 'A' CCW pump is blocked since 'C' CCW pump is already running
B. Incorrect – this would cause an autostart of the B CCW not the A CCW pump
C. Incorrect – discharge pressure would have to lower less than 28.65 psig for longer than 4 seconds.
D. Correct – In this situation, NB01 would lose power and when the EDG starts and closes on NB01, generating a shutdown sequencer, the A CCW pump would start at the 5 second time

Technical Reference(s):
1. M-22EG01, CCW P&ID, Rev 11

2. OTA-RK-00020, ADD 52B, Rev 0
3. E-23EG01C, B CCW Pump Schematic Diagram, Rev 9
4. OTN-EG-00001, CCW system, Rev 61
5. E-22NF01, Load Shedding and Emergency Load Sequence Logic, Rev 8

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #10, Objective C: DESCRIBE the purpose and operation of the following CCW System components:
1. CCW Pumps

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Pressurizer Pressure Control	Group #	1		
	K/A #	010 A2.02		
	Importance Rating	3.9		
Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Spray valve failures				

Question # 36

Reactor Power is 40%.

- Pressurizer Spray Valve, BB PCV0455B, FAILS OPEN and can NOT be manually closed
- Pressurizer Pressure begins to slowly lower

(1) What is the LOWEST pressure that the Pressurizer Backup Heaters should automatically energize?

And

(2) If Pressurizer Pressure continues to lower after the Pressurizer Backup Heaters are energized, the crew should stop which RCPs per OTO-BB-00006, Pressurizer Pressure Control Malfunction?

A. (1) 2210 psig
(2) A and D

B. (1) 2210 psig
(2) B and D

C. (1) 2220 psig
(2) A and D

D. (1) 2220 psig
(2) B and D

Answer: A

Explanation:

OTO-BB-00006 will be performed and at step #20, when both PZR Sprays valves are checked closed, the RNO action will be performed. In this RNO as pressure continues to lower after the backup heaters are on, a reactor and turbine trip are required along with securing the appropriate RCPs: For BB PCV455B it is the A and D RCP. The distractors of the B and D RCP are for BB PCV455C.

The variable heater would be fully energized at 2220 psig and the backup heaters would be on @2210 psig

A. Correct – See above explanation
B. Incorrect – wrong RCPs
C. Incorrect – Wrong pressure. This is when variable heaters are fully energized
D. Incorrect – both are wrong

Technical Reference(s):

1. OTO-BB-00006, Pressurizer Pressure Control Malfunction, Rev 20
2. OTN-BB-00005, Attachment 1, Master Pressure Controller, Rev 15

References to be provided to applicants during examination: None

Learning Objective:
T61.003B, Off Normal Operations, LP #41, OTO-BB-00006, Objective C: Given a set of plant conditions or parameters indicating a Pressurizer Pressure Control Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

T61.0110, Systems, LP #9, RCS, Objective B: DESCRIBE the purpose and operation of the following RCS components to include interlocks, controller operations and power supply:

5. Power Operated Relief Valves (PORVs)

Question Source: Bank # __X – no bank id____
Modified Bank # ______
New _______

Question History: Last NRC Exam _____2016 ILT NRC Exam Q#36_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

k/a match as the impact of a spray valve failures in PZR PCS will be that PZR Pressure will lower. There are no other plausible PZR Pressure responses. The impacts of this failure are the automatic system response (backup heater automatically turning on) and if this operation is not successful in stopping PZR Pressure from lowering, stopping the required RCPs is directed by the off normal procedure to mitigate the spray valve failure. The impacts and interrelationships of PZR PCS and RPS are tested in the previous question so to prevent overlap, RPS impacts are N/A for this question.

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier	2		
Reactor Protection	**Group**	1		
	K/A	012 K1.06		
	Importance Rating	3.1		
Knowledge of the physical connections and/or cause-effect relationships between the RPS and the following: T/G				

Question # 37

Reactor Power is at 42% performing a plant shutdown per OTG-ZZ-00004, Power Operations.

As turbine power is lowered below 42% power, AC-PT-505, Turbine First Stage Impulse Pressure detector, fails AS IS at 42%.

Assuming NO operator action(s) and the power descension continues, what correctly describes the effect of the AC-PT-505 failure?

A. The reactor will trip if B RCP trips at 15% power

B. The SR NIs will have to be manually unblocked below P6

C. An automatic reactor trip will occur when the turbine is tripped at 15% power

D. The reactor will automatically trip if PZR pressure lowers to 1860 psig with ALL PR NI channels at 8%

Answer: D

Explanation:

When AC-PT-505 fails at 42%, permissive P13, turbine at power, will prevent P7 from blocking the six low power reactor trips. P7 is satisfied when both P10 and P13 are below their setpoints.

A is incorrect. P8 setpoint is 48% NI power, which is not affected by the AC-PT-505 failure
B is incorrect, P10, which is based on 10% NI power, will not prevent the SR channels from energizing.
C is incorrect. P-9 setpoint is 50% NI power, which is not affected by the AC-PT-505 failure .
D is Correct P7 will not block the low power reactor trips. The reactor would trip if PZR pressure dropped to below the Rx trip setpoint but remained above the SI setpoint.

Technical Reference(s):
1. OTO-SA-00001 ESFAS Verification and Restoration, Rev 41

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References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #27, Objective C: LIST all the Reactor Trip Signals supplied to RPS, including setpoint, coincidence, interlocks and protection afforded.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier	2		
Reactor Protection	**Group**	1		
	K/A	012 A3.02		
	Importance Rating	3.6		
Ability to monitor automatic operation of the RPS, including the following: Bistables				

Question # 38

With Reactor Power at 15%, what trip bistable is blocked?

A. Pressurizer low pressure

B. Pressurizer high level

C. Low flow in a single loop

D. Reactor coolant pump undervoltage

Answer: C

Explanation:

A: INCORRECT: The Pressurizer low pressure is blocked at less than 10% power.
B: INCORRECT: The Pressurizer high level is blocked at less than 10% power.
C: CORRECT: Low flow in a single loop is blocked at less than 48% power.
D: INCORRECT: Reactor coolant pump undervoltage is blocked at less than 10% power.

Technical Reference(s):
1. Technical Specification 3.3.1, RPS Instrumentation

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP#27, Objective C: LIST all the Reactor Trip Signals supplied to RPS, including setpoint, coincidence, interlocks and protection afforded.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

Question Cognitive Level:

Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier	2		
Engineered Safety Features Actuation	**Group**	1		
	K/A	013 A4.03		
	Importance Rating	4.5		
Ability to manually operate and/or monitor in the control room: ESFAS initiation				

Question # 39

Ten minutes after a LOCA, containment pressure indicates the following:

- PT-934 = 26.8 psig
- PT-935 = 27.2 psig
- PT-936 = 27.1 psig
- PT-937 = 26.9 psig

What describes the response of the Containment Spray System 10 seconds after containment pressure reaches the listed values? (Assume bistables actuate at their exact setpoint and no ESF systems have been reset)

A. ONLY the pumps have started

B. the pumps have started and their discharge valves have started to open

C. ONLY the pump discharge valves have started to open

D. the pumps have NOT started and the discharge valves have NOT started to open

Answer: B

Explanation:
If a high Containment pressure condition exists, a Safety Injection Signal and a CIS A will be initiated at 3.5 psig containment pressure.
- *The SIS will cause a LOCA sequencer actuation.*
- *Loads will be sequenced onto the NB buses at their designated starting times*

At time #3 (15 seconds on the LOCA Sequencer), if pressure in containment is > 27 psig, a CSAS will be present, and the containment spray pumps will be started. If pressure in containment is < 27 psig, a CSAS will not be present and the spray pumps will not be started. If a CSAS is not present at time 3 (15 seconds), a 26.5 (+/-1.5 secs)second time delay is initiated. After the 26.5second time delay, the spray pump will start if a CSAS occurs. This is to prevent the spray pump from starting at the same time another load is being sequenced onto the NB bus.

A INCORRECT: The valves directly open from the actuation signal.
B: CORRECT: Containment Spray actuates on a 2/4 logic at 27.0 psig. The valves directly open from the actuation signal. The pumps will start providing there is still a Sequencer signal present which locks in a 40 seconds after the sequencer started. The sequencer started on the initial Safety Injection. (CTMT pressure at 3.5 psig)
C: INCORRECT: The pumps will also start because the sequencer has not been reset.
D: INCORRECT: The valves open directly from the actuation signal and the pumps will start from the sequencer.

Technical Reference(s):
1. OTA RK-00020 ADD 59A, Containment Spray Actuation Signal, Rev 0
2. OSP-EN-V001A, TRAIN A CONTAINMENT SPRAY VALVE OPERABILITY, Rev 22
3. OSP-EN-V001B, TRAIN B CONTAINMENT SPRAY VALVE OPERABILITY, Rev 23

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #51, Objective C: DISCUSS the reason for the five (5) second sequencing interval and LIST the major loads sequenced by the following:
1. LOCA Sequencer.

Question Source: Bank # _
Modified Bank # _
New ___X___

Question History: Last NRC Exam __N/A__

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Containment Cooling	**Group**	1		
	K/A	022 K4.02		
	Importance Rating	3.1		
Knowledge of the design feature(s) and/or interlocks which provide for the following: Correlation of fan speed and flowpath changes with containment pressure				

Question # 40

Why do containment coolers start in slow speed when a valid SI signal due to Containment pressure is present?

A. Limits the electrical load on the emergency diesels due to the denser atmosphere in containment.

B. Limits the amperage the fans draw due to the denser atmosphere in containment.

C. Maintain the cooler heat balance since the cooling flow will ~ double.

D. Maintain cooler heat balance when containment spray starts

Answer: B

Explanation:

A: Incorrect.- Plausible as it would limit the KW usage and therefore load on EDG but is not the reason
B: CORRECT – the reason is to protect the motors by limiting the amperage
C: Incorrect – Plausible as flow will double and correct per thermodynamics but is not the reason
D: Incorrect – Plausible since the cooler fans work in combination with containment spray to lower containment pressure

Technical Reference(s):
1. OTN-GN-00001, Containment Cooling and CRDM Cooling, Rev 29

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #40, Objective B. DESCRIBE the purpose and operation of the following containment cooling system components.
1. Containment Fan Coolers

Question Source: Bank # X L5587_
Modified Bank # _____
New ______

Question History: Last NRC Exam ____N/A___

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

K/A match as the question ask about the design features and reasons for cooler fan speed when a higher containment pressure is present. Containment Cooling air flowpath does not change based on containment pressure. Containment cooler cooling water course does change based on containment pressure but is examined on a different question.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Containment Spray	**Group #**	1		
	K/A #	026 K1.01		
	Importance Rating	4.2		
Knowledge of the physical connections and/or cause/effect relationships between the CSS and the following systems: ECCS				

Question # 41

What pumps can be lined up to take a DIRECT suction from the containment recirculation sump?

A. RHR and SI

B. RHR and Containment Spray

C. CCPs and Containment Spray

D. CCPs and SI

Answer: B

Explanation:

Both the RHR and Containment Spray pumps can be lined up to take a suction directly from the containment recirculation sump. See M22 references. Other ECCS pumps can be lined up to take an INDIRECT suction from the containment sumps during the performance of several Emergency procedures. See explanation below.

A. Incorrect - The RHR pumps can be lined up to directly take a suction from the containment recirculation sump. The SI pumps cannot be directly lined up to take a suction from the containment recirculation sump. Plausible because when the RWST is low and the SI pumps are used the SI pumps take a suction on the RHR header that is lined up to the containment recirculation sump.
B. Correct
C. Incorrect - The CCPs cannot be directly lined up to take a suction from the containment recirculation sump. Containment Spray pumps can be lined up to take a suction directly from the containment recirculation sump. Plausible because when the RWST is low the CCPs can be lined up to take a suction on the RHR header that is lined up to the containment recirculation sump. Additionally, the CCP suctions are realigned during the performance of emergency procedures and the candidate may falsely believe / remember that during the performance of these the CCP is aligned to the containment sump. Specifically, in ECA 1.1 Sump Blockage Mitigation, there are several actions and steps to verify CCPs show no signs cavitation and if there is cavitation the operator is directed to secure the pump. The operator could falsely believe that the CCPs can be directly aligned to the sump by the actions provided in this procedure. (i.e. the pump is cavitation

when performing actions to mitigate sump blockage, I must secure the pump as its suction source is blocked).
D. Incorrect – see above explanations for CCP and SI pumps.

Technical Reference(s):
1. M-22EJ01, P&ID RHR System, Rev 62
2. M-22EN01, P&ID Containment Spray System, Rev 16

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #18, Containment Spray, Objective B: DESCRIBE the purpose, operation and location of the following Containment Spray System components:
1. Containment Spray Pumps
2. Containment Recirculation Sump

Question Source: Bank # __X – no bank id____
Modified Bank # ______
New _______

Question History: Last NRC Exam _____2016 ILT NRC Exam – Q#41_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Containment Spray	**Group #**	1		
	K/A #	026 K2.01		
	Importance Rating	3.4		
Knowledge of bus power supplies to the following: Containment spray pumps				

Question # 42

What is the power supply to the 'B' Containment Spray Pump, DPEN01B?

A. NG01

B. NG02

C. NB01

D. NB02

Answer: D

Explanation:

Per the reference Table B, the power supply to the motor is NB0203. All of the above choices are safety related AC busses. The NG busses are 480 VAC and the NBs are 4160 VAC. NG01 and NB01 are A Train power supplies while NG02 and NB02 are B Train.

A. Incorrect but plausible if the candidate believes it is a 480 VAC motor powered from an A Train MCC.
B. Incorrect but plausible if the candidate recalls that it is the correct safety train but believes it is a 480 VAC motor.
C. Incorrect but plausible if the candidate recalls that the motor is 4160 VAC but wrong as this is the A Train power supply.
D. Correct

Technical Reference(s):
1. E-23EN01A, Schematic Diagram Containment Spray Pump B DPEN01B, Rev 0

References to be provided to applicants during examination: None

Learning Objective: T61.0110 systems, LP #18, Objective B: DESCRIBE the purpose, operation and location of the following Containment Spray System components:
1.Containment Spray Pumps

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Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(8)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Main and Reheat Steam	Group	1		
	K/A	039 K5.01		
	Importance Rating	2.9		
Knowledge of the operational implications of the following concepts as they apply to MRSS: Definition and causes of steam/water hammer				

Question # 43

A plant heatup is in progress in accordance with step 5.7.4 (Initiate main steam warmup) of OTG-ZZ-00001, Plant Heatup From Cold Shutdown to Hot Standby.

What is the next action required, and the reason for that action, in accordance with OTG-ZZ-00001?

A. Periodically cycle low point steam drains during the heatup to prevent water hammer in the main steam lines

B. Isolate main steam drains when RCS temperature is above 300°F to prevent water hammer in the main steam lines

C. Maintain main steam drains open when RCS temperature is above 300°F to expedite RCS heatup

D. Maintain steam trap bypass valves open during the heatup to expedite RCS heatup

Answer: A

Explanation:
A: CORRECT: - Step 5.7.5 of OTG-ZZ-00001(During RCS Heatup INITIATE Step 5.7.5.a OR 5.7.5.b to prevent water hammer in the Main Steam lines):
B: INCORRECT - because water buildup may water hammer if valves are closed.
C: INCORRECT -:Note at step 5.7.5 says In order to expedite RCS Heatup, Secondary Plant steam drains and steam trap bypasses may be periodically cycled per Attachment 7, Steam Load Management For RCS temperature Control instead of continuously open as instructed I n the Normal Operating Procedures because maintaining valves open will delay the heatup.
D: INCORRECT - Note at step 5.7.5 says In order to expedite RCS Heatup, Secondary Plant steam drains and steam trap bypasses may be periodically cycled per Attachment 7, Steam Load Management For RCS temperature Control instead of continuously open as instructed I n the Normal Operating Procedures because maintaining valves open will delay the heatup

Technical Reference(s):
1. OTG-ZZ-00001, Plant Heatup From Cold Shutdown to Hot Standby, Rev 091

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References to be provided to applicants during examination: None

Learning Objective: T61.GFES, LP #27, Objective #43: Explain operational implications of water (fluid) hammer.

T61.0110 Systems: LP #31, Objective I: DISCUSS the following operating conditions associated with the Main Turbine and Auxiliary Components:

1. Initial Conditions to Admit Steam
2. Turbine Shell Warming
3. Steam Chest Warming
4. Turbine Roll

Question Source: Bank # ____16251_____
Modified Bank # _____
New _______

Question History: Last NRC Exam ___N/A____

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(14)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Main Feedwater	**Group #**	1		
	K/A #	059K3.03		
	Importance Rating	3.5		
Knowledge of the effect that a loss or malfunction of the MFW will have on the following: S/Gs.				

Question # 44

Reactor Power is 100%.

Which Main Feedwater malfunction will cause SG water level to rise and then return to its original value?

A. ‘A’ MFRV Bypass valve fails open

B. SG ‘C’ Feed Flow Channel, AE FT-530, fails high

C. ‘B’ Main Feedwater Pump recirculation valve fails open

D. Main Feed Header Pressure Channel, AE PT-508, fails high

Answer: A

Explanation:

A. Correct, If the bypass fails open SGWL will initially trend up until the SGWLC system takes control and throttles the MFRWs close.
B. Incorrect, If the feed flow signal is not good quality it is not used to determine feed flow to the SG. It then does not change SGWL
C. Incorrect, If the B MFP recirc valve were to fail open less water would go to the SG initially and SGWL would lower until the SGWL control system would compensate and raise level
D. Incorrect, If the feed header pressure were to fail high it would cause the MFPs to slow down which would cause SGWL to go down

Technical Reference(s):.
1. OTO-AE-00002, SGWL Control Malfunctions, Rev 13
2. OTO-AE-00001, Feedwater System Malfunctions, Rev 37

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP-23, MAIN FEEDWATER SYSTEM – AE,
Obj E DESCRIBE the operation, including signal inputs, of the MFW pump speed control

system and EXPLAIN the control response to input failures.

Obj F DESCRIBE the operation, including signal inputs, of the MFW REG valves and EXPLAIN the control response to input failures.

Obj G DESCRIBE the operation, including signal inputs, of the MFW REG bypass valves and EXPLAIN the control response to input failures.

Obj H. IDENTIFY the MFW MCB controls, indications, and alarms and DESCRIBE how each is used to monitor and/or control the MFW System.

Question Source: Bank # ___X_2016 Audit Exam Q#44__
Modified Bank # ______
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Auxiliary /Emergency Feedwater	**Group**	1		
	K/A	061 A2.03		
	Importance Rating	3.1		
Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of DC				

Question # 45

A Reactor trip occurs from 100% due to a bus fault on NK04.

Normal control power to the ___(1)____ is lost. The crew should utilize ____(2)____ for control of AFW flow during these conditions.

A. (1) 'B' MDAFP
   (2) ES-0.1, Reactor Trip Response

B. (1) 'B' MDAFP
   (2) OTO-NK-00002, Loss of Vital 125 VDC Bus

C. (1) TDAFP
   (2) ES-0.1, Reactor Trip Response

D. (1) TDAFP
   (2) OTO-NK-00002, Loss of Vital 125 VDC Bus

Answer: A

Explanation:

A: CORRECT: NK04 is B train and ES-0.1 provides guidance on AFW flow
B: INCORRECT: OTO-NK-00002 does not provides guidance on AFW flow
C: INCORRECT: NK04 has no impact on TDAFP
D: INCORRECT: OTO-NK-00002 does not provides guidance on AFW flow

Technical Reference(s):
1. OTO-NK-00002 Loss of Vital 125 VDC Bus, Rev 16
2. ES-0.1 Reactor Trip Response, Rev 18

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #25 Objective D: IDENTIFY the AFW Main Control

Board (MCB) and local controls, alarms and indications and EXPLAIN how each is used to predict, monitor or control the AFW system.

Question Source: Bank # _L16717__
Modified Bank # ______
New ______

Question History: Last NRC Exam ____2009________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
AC Electrical Distribution	**Group**	1		
	K/A	062 A4.03		
	Importance Rating	2.8		
Ability to manually operate and/or monitor in the control room: Synchroscope including an understanding of running and incoming voltages				

Question # 46

The main generator is being paralled to the grid. The RO has adjusted generator voltage and the synchroscope direction. The BOP verifies that generator voltage is ___(1)___ than grid voltage and the sychroscope is rotating slowly in ___(2)___ direction prior to RO correctly synching on main generator.

A. (1) higher
   (2) clockwise

B. (1) higher
   (2) counterclockwise

C. (1) lower
   (2) clockwise

D. (1) lower
   (2) counterclockwise

Answer: A

Explanation:

A CORRECT: per OTN-AC-00001 steps 5.6.11 and 5.6.15.
B: INCORRECT: synchroscope rotating in wrong direction
C: INCORRECT: generator voltage required to be higher than grid voltage
D: INCORRECT: synchroscope rotating in wrong direction.

Technical Reference(s):
1. OTN-AC-00001, Main Turbine and Generator Systems, Rev 51

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #34, Objective D: EXPLAIN how to synchronize the Main Generator onto the Grid.

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Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	2		
DC Electrical Distribution	**Group #**	1		
	K/A #	063 K2.01		
	Importance Rating	2.9		
Knowledge of bus power supplies to the following: Major DC loads				

Question # 47

What is the power supply to the Emergency Seal Oil Pump, DPCD003?

A. NK01

B. NG01

C. PK01

D. PJ01

Answer: D

Explanation:

The non-safety related 250VDC system (PJ01) supplies power to emergency DC oil pumps. The specific loads supplied by the system include:
* *turbine generator emergency lube oil pump*
* *generator emergency seal oil pump*
* *two main feedwater turbine emergency lube oil pumps*

The non-safety related 125VDC power system (PK) has 5 buses, PK01 - PK05. Each bus has a battery charger and a battery. The function of PK01-PK04 is to supply non-safety related instrumentation and control loads required for power generation. PK05 supplies control power to PB05 breakers.

The Distractors of NK01 and PK01 support each other in terms of plausibility as the candidate must decide if it is a safety related or nonsafety related 125 VDC supply. Likewise the 2 safety related choices (NK01 and NG01) and 2 nonsafety (PK01 and PJ01) related choices support the plausibility and distribution of possible answers/choices.

A. Incorrect but plausible if the candidate believes it is safety related DC load and believes it is a 125 VDC load (as there isn't a safety related 250 VDC bus)
B. Incorrect but plausible if the candidate believes it is safety related and knows it is 'more than 125VDC' with the next available safety related bus being a 480 VAC choice, i.e. NG01
C. Incorrect but plausible as it is a non safety related bus but wrong as PK01 is a 125 VDC bus not 250 VDC
D. Correct

Technical Reference(s):
1. E-21PJ01, 250 Volt DC System, Rev 4

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #2, Service Power, Objective F: IDENTIFY the loads supplied by the following busses, which busses can be cross-tied, and DESCRIBE the consequences of losing each bus:
1. PA01
2. PA02
3. PG Busses
4. PB03
5. PB04
6. PJ01

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(4)

Comments:

k/a match as it test the power supply to a large DC load aka the 20hp generator emergency seal oil pump which is important as it prevents hydrogen leakage onto the turbine deck and associated hazardous conditions. This was also choosen to minimize overall with other K2 question on this exam.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
063 DC Electrical Distribution	Group #	1		
	K/A #	00063 A1.01		
	Importance Rating	2.5		
Ability to predict and / or monitor changes in parameters associated with operating the DC electrical system controls including: Battery capacity as it is affected by discharge rate.				

Question # 48

A Loss of All AC Power has occurred. NK14 battery ammeter indicates a 200 amp discharge.

What is the MAXIMUM time that NK14 could supply NK04 bus? (Assume NK14 was fully charged at the time of the loss of all AC power.)

A. 4.5 hours

B. 8 hours

C. 9 hours

D. 12 hours

Answer: B

Explanation:
Per OTO-NB-00002, the capacity and calculated times of the NK batteries are as follows:
NK02 = 900 amp hours= 900/200 = 4.5 hours
NK04 = 1650 amp hours = 1650/200 = 8.25 hours
PK02 = 2400 amp hours = 2400/200 = 12 hours

A. Incorrect – plausible if confusing NK02 amp hour rating
B. Correct
C. Incorrect – plausible because if the student uses the correct 1650 rating but does not understand that the battery would be dead at the 8.25 hours time therefore it could not supply it for 9 hours
D. Incorrect - plausible if confusing PK02 amp hour rating

Technical Reference(s):.
1. OTO-NB-00002, Loss of Power to NB02, Rev 31, Step #25

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References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #9 Safeguards Power, Objective M: EXPLAIN the precautions, limitations and bases for the following components/conditions associated with OTN-NK-00001, "Class 1E 125 VDC Electrical System":

1. Battery capacity
2. Maximum NK Battery Charge amperage output

Question Source: Bank # ___X R12275___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Emergency Diesel Generator	Group #	1		
	K/A #	064 K6.07		
	Importance Rating	2.7		
Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers.				

Question # 49

Reactor Power is 100%.

- A relief valve has failed open on 'C' Starting Air Receiver for 'B' Emergency Diesel Generator (EDG).
- The leakage exceeds the capacity of the starting air compressors.

If an EDG start signal is generated, the ‘B’ EDG ___(1)___ start, because the ‘C’ and ‘D’ air receivers ____(2)____ cross connected.

	(1)	(2)
A.	will	are
B.	will	are not
C.	will not	are
D.	will not	are not

Answer: B

Explanation: *The starting air compressor system for the B EDG consists of 2 Starting air tanks TKJ02C and TKJ02D. These are refered to as the C and D starting air receivers. Each Starting air tank has an inlet check valve, KJV711B and KJV712B, that prevents a flaw / depressurization in one air tank from affecting the other tank. The discharge of the starting air tanks are NOT cross connected, i.e. KJV760B is closed.*

A. Incorrect – Plausible if the candidate assumes that the air pressure in the D receiver will compensate for the loss of the C air receiver and the receivers are cross connected. The EDG will start because the Air receivers ARE NOT cross connected therefore the fault in the C air receiver will not affect the D air receiver allowing the EDG to start on only one receiver.
B. Correct – the D Air receiver has sufficient starting air pressure required for more than one start attempt per the Tech Spec basis. Prior to leak on the C Air receiver, it is assumed that starting receiver pressure was 610- 640 psig (normal band) The Air receivers are not cross connected during normal operation to ensure redundancy in the starting capability of the EDG. The inlet lines

have check valves which allow a single air compressor to supply both receivers simultaneously while still ensuring independence and redundancy of the system.
C. Incorrect. Plausible if the candidate assumes that both air receivers are required to supply sufficient starting air to the EDG with the system operated cross connected. If they assume they are operated cross connected they could also assume the fault in the C receiver is also degrading the starting pressure in the D receiver.
D. Incorrect. Plausible if the candidate assumes that both air receivers are required to supply sufficient starting air to the EDG with the system operated without the receivers cross connected.

Technical Reference(s):
1. M-22KJ02, P&ID Standby Diesel Generator "A" Intake Exhaust, F.O. &Start Air System, Rev 20

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP#3 KJ NE Standby Generation Objective C: Describe the purpose, major components and operation of the following Standby Diesel Generator support systems: Air Start System

Question Source: Bank # ___X__L16192_
Modified Bank # ______
New _______

Question History: Last NRC Exam __2014 ILT NRC Exam Q#49__________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Emergency Diesel Generator	Group #	1		
	K/A #	064 K3.02		
	Importance Rating	4.2		
Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following: ESFAS controlled or actuated systems.				

Question # 50

The Plant was operating at 100% power when a LOCA occurred in conjunction with a Loss of Off-site Power.

Current plant conditions are as follows:

- E-1, Loss of Reactor or Secondary Coolant, is in progress
- ‘B’ Emergency Diesel Generator NE02 has TRIPPED
- Containment pressure peaked at 28 psig and is now lowering
- ANN 47B, RWST LEV LOLO 2, is in alarm

What describes the position of EN HV-1 (Train A) and EN HV-7 (Train B), Containment Recirc Sump to Containment Spray Pumps, at the completion of ES-1.3, Transfer To Cold Leg Recirculation?

	EN HV-1	EN HV-7
A.	OPEN	OPEN
B.	OPEN	CLOSED
C.	CLOSED	OPEN
D.	CLOSED	CLOSED

Answer: B

Explanation:

A. Incorrect. Plausible because under normal conditions with power available to both A and B train components, this would be correct. With the loss of NB02 due to the trip of the EDG, the B Train valves (including EN HV-7) and Spray Pump will not have power.

B. Correct. ES-1.3 will direct the operator to open both the containment sump recirculation valves. With the loss of NB02 due to the trip of the EDG, the B Train valves (including EN HV-7) and Spray Pump will not have power.

C. Incorrect. Plausible if the candidate incorrectly crosses the trains believing that the A train components are affected by the loss of the EDG. Incorrect because EN HV-1 (A train) will be aligned correctly to the Open position, and EN HV-7 (B Train) will not be realigned.
D. Incorrect. Plausible if the candidate determines that containment pressure is below the setpoint for containment spray actuation and therefore the spray pumps are not running and swap over for containment spray is not required. Incorrect because Containment spray actuates at 27 psig in containment and swapover to containment sump is required at the RWST Lo Lo 2 alarm setpoint.

Technical Reference(s):
1. ES-1.3, Transfer to Cold Leg Recirculation, Rev 13
2. E-23EN02, Schematic Diagram, Containment Recirc Sump Isolation Valves, Rev 14

References to be provided to applicants during examination: None.

Learning Objective: T61.0110, Systems, LP#18, Obj. C. Explain the interlocks, controls and power supplies to:
1) Containment Spray Pumps,
2) Containment Recirculation Sump Encapsulated Suction Valves,
3) Containment Spray Pump Discharge Valves.

Question Source: Bank # __x – no bank id ____
Modified Bank # _____
New _______

Question History: Last NRC Exam ______2014 ILT NRC Exam Q#8______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

K/A Match: Candidate must know that the Loss of the Emergency Diesel Generator results in a loss of the Vital Power Bus, and how the of the loss of the power to the bus will impact the ESFAS actuated Containment Spray System.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Process Radiation Monitoring	Group	1		
	K/A	073 G2.1.23		
	Importance Rating	4.3		
Ability to perform specific system and integrated plant procedures during all modes of plant operation				

Question # 51

The Reactor Operator observing process radiation monitor control panel RM-11 discovers GK RE-04 is displayed in MAGENTA.

What action is required?

A. Perform surveillance checks every 30 minutes

B. Restore the monitor using the RM-23 panel controls

C. Restore flow using the controls on the RM-11 panel

D. When purge time is complete verify green OPERATE light LIT on RM-23

Answer: A

Explanation:

A Correct - OTA-SP-RM011 states: "RM-11 Communication Failure Magenta" Attachment 43 states: IF the monitor is TECH SPEC or a FSAR 16.0 CHAPTER Monitor with RM-23, PERFORM the following: (See Step 4.b for a list of monitors for this category) 1) ESTABLISH a 30 minute surveillance for the monitor by observing the RM-23.

B Incorrect - GK RE-04 has been taken off line and is INOPERABLE. Restore the monitor using the RM-23 panel controls. - Plausible since this a misinterpretation of color coding and RM 23 would be used to restore monitor functions.

C Incorrect - GK RE-04 has lost process flow and is INOPERABLE. Restore flow using the controls on the RM-11 panel. - Plausible since the RM-11 has controls to operate various monitors, however loss of flow causes a dark blue system failure alarm.

D Incorrect - GK RE-04 is being purged, but is OPERABLE. When purge time is complete verify green OPERATE light LIT on RM-23. - Plausible since this is misinterpretation of color code and the operate light validates operability.

Technical Reference(s):

1. OTN-SP-00002, RM11 Console Operations, Rev 008
2. OTA-SP-RM011, Radiation Monitor control panel RM11, Rev 043

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References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP#36, Objective C. IDENTIFY the Process and Area Radiation Monitoring Control Room controls, alarms, and indications and DESCRIBE how each is used to predict, monitor and control the Process and Area Radiation Monitoring System.

Question Source: Bank # _23269..__
Modified Bank # ______
New ______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Service Water	Group	1		
	K/A	076 K3.07		
	Importance Rating	3.7		
Knowledge of the effect that a loss or malfunctions of the SW system will have on the following: ESF loads				

Question # 52

Reactor Power is 100%.

- 'C' Service Water Pump is OOS for maintenance
- 'A' and 'B' Service Water Pumps subsquently trip

(1) What signal starts ESW Pumps

And

(2) What loads are being cooled by ESW?

A. (1) Automatic
(2) Safety related only

B. (1) Automatic
(2) Safety and non-safety related

C. (1) Manual
(2) Safety related only

D. (1) Manual
(2) Safety and non-safety related

Answer: C

Explanation:
Pump controls consist of two hand switches per pump with pull to lock, stop and run positions. One switch is located locally in the ESW pump room and the other is located on the main control board. The pump can be started manually by either associated handswitch or automatically by any of the following train related signals:

- *Shutdown sequencer actuation*
- *LOCA sequencer actuation*
- *Aux. Feedwater low suction pressure with an AFAS present*
- *Undervoltage on opposite train NB Bus with low ESW flow to Containment Coolers on its*

train.
Check valves in Service Water lines prevent ESW from supplying non–safety related loads

-

A: INCORRECT ESW pumps do not auto start on low service water pressure
B: INCORRECT ESW pumps do not auto start on low service water pressure
C: CORRECT ESW pumps must be started manually and supply only ESF Loads
D: INCORRECT ESW only supplies ESF loads since check valve in supply line prevents ESW from cooling non-safety related loads

Technical Reference(s):
1. M-22EF02, P&ID Essential Service Water System, Rev 77
2. M-22EF01, P&ID Essential Service Water System, Rev 81
3. E-22NF01, Load Shedding and Emergency Load Sequencing Logic, Rev 8
4. E-U3EF01, Schematic Diagram Essential Service Water Pump A, Rev 032

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #05, Objective D: DESCRIBE the operation of the Essential Service Water System under the following conditions:
1. Standby
2. Safety Injection Signal
3. Loss of Offsite Power
4. Low Suction Pressure to the Auxiliary Feedwater Pumps
5. Opposite train NB bus undervoltage with low flow to the CTMT coolers.

Question Source: Bank # ______
Modified Bank # ______
New ___X___

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Instrument Air	Group #	1		
	K/A #	078 K1.04		
	Importance Rating	2.6		
Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Cooling water to compressor				

Question # 53

The plant was tripped due to a LOCA

- KAHA0043, Compress Air SYS AIR CMPSRS Sequence Selector SW, is in the C-A-B position
- 'A' and 'C' air compressors are in REMOTE
- 'B' air compressor is in LOCAL
- A leak causes air pressure to begin to lower from the normal operating pressure
- Air pressure is currently 118 psig and lowering slowly due to an air leak

(1) Which air compressor is currently loaded?

And

(2) Which air compressor(s) is/are being cooled by Essential Service Water?

A. (1) 'C'
(2) 'C'

B. (1) 'C'
(2) 'A' and 'B'

C. (1) 'A'
(2) 'C'

D. (1) 'A'
(2) 'A' and 'B'

Answer: B

Explanation:

The Air Compressors automatically sequence as follows:

Compressor	***Load (psig)***	***Unload (psig)***
Lead	*119*	*128*

1st Backup	*117*	*126*
2nd Backup	*115*	*124*

So given the air pressure of 118 psig, KAHS0043 switch position of lead/lag/lag-lag of C-A-B and B being in LOCAL, only C will be loaded

A. Incorrect C is cooled by chilled water system
B. Correct – A and B are cooled by ESW in this condition and C is only compressor loaded at 118 psig
C. Incorrect – A(i.e. the lag compressor) is not loaded at 118 psig and C is cooled by chilled water system
D. Incorrect – A(i.e. the lag compressor) is not loaded at 118 psig

Technical Reference(s):
1. OTO-KA-00001, Partial or Total Loss of Instrument Air, Rev 28
2. OTN-KA-00001, Compressed Air System, Rev 27
3. M-22KA01, PIPING & INSTRUMENTATION DIAGRAM - COMPRESSED AIR SYSTEM
4. FSAR FIGURE 9.3-1 SHEET 1, Rev 035

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #14, Objective B, E &F:

B. DESCRIBE the purpose and operation of the following Service and Instrument Air components:
1. Air Compressors
2. Air Dryers
3. N2 Accumulators
4. Flow restricting orifices
5. Sequential Loading Panel
6. Pressure Control Valve (KA-PV-11)
7. Containment Instrument Air Isolation Valve (KA-FV-29)

E. DESCRIBE the actions that occur as air pressure falls from 120 to 100 psig.

F. IDENTIFY the Service and Instrument Air System Main Control Board (MCB) controls, alarms and indications and DESCRIBE how each is used to predict, monitor or control changes in the Service and Instrument Air System

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

Question Cognitive Level:
Memory or Fundamental Knowledge _____

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Comprehension or Analysis __X___

10 CFR Part 55 Content:

10 CFR 55.41(b)(4)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier	2		
Containment	**Group**	1		
	K/A	103 A1.01		
	Importance Rating	3.7		
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating containment systems controls including: Containment pressure, temperature, and humidity				

Question # 54

Reactor Power is 100%.

- There is a LARGE Secondary Steam Leak INSIDE containment
- Containment Pressure is 1.5 psig and rising
- Containment Temperature and Humidity are rising

What automatic actions will occur to stop the containment pressure and temperature rise? Assume no operator actions.

A. Reactor Trip

B. Safety Injection Signal

C. Steam Line Isolation Signal

D. Containment Spray Actuation Signal

Answer: D

Explanation:

Incorrect - Reactor Trip may slow, but not stop rise in containment pressure and temperature.
Incorrect - Safety Injection may also slow, but not stop rise in containment pressure and temperature.
Incorrect - Steam Line Isolation Signal may increase rate of rise in containment pressure and temperature due to Steam Generator pressure increase.
Correct - Containment Spray will lower containment temperature and pressure.

Technical Reference(s):
1. OTA-RK-00020, Addendum 59A, Containment Spray Actuation Signal, Rev 000
2. T.S. 3.3.2 Basis

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP#18, Objective G DESCRIBE those conditions that will generate a containment spray actuation signal (CSAS) and IDENTIFY the Containment Spray System response to a CSAS.

Question Source: Bank # __X____
Modified Bank # ______
New ______

Question History: Last NRC Exam ___2011___

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	2		
Containment	**Group #**	1		
	K/A #	103 A4.04		
	Importance Rating	3.5		
Ability to manually operate and/or monitor in the control room: Phase A and phase B resets				

Question # 55

In order to reset a valid Phase B (CIS B) signal, what are the MINIMUM action(s)/plant condition(s) required?

A. Push reset for CIS B ONLY

B. Push resets for both CSAS and CIS B ONLY

C. Reduce Containment Pressure less than CTMT HI 3 and push reset for CIS B ONLY

D. Reduce Containment Pressure less than CTMT HI 3 and push resets for both CSAS and CIS B

Answer: A

Explanation:

A CISB signal can be reset at any time following its actuation without having to reset any other signal. CTMT HI 3 is an actuation signal for CSAS and CISB (see reference).

A. Correct – See above explanation.
B. Incorrect – CISB may be reset without resetting CSAS but plausible as both are initiated by the same pressure setpoint and logic.
C. Incorrect – CISB may be reset without clearing of CTMT HI 3 signal.
D. Incorrect – CISB may be reset without resetting CSAS and clearing of CTMT HI 3 signal

Technical Reference(s):
1. 7250D64 S008 - Functional Diagram Safeguards Actuation Signals, Rev 4
2. OTO-SA-00001, ESFAS Verification and Restoration, Rev 41, Attachment C and D,

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #27 Reactor Protection, Objective F: LIST the

Engineered Safety Features (ESF) actuations that may be generated by the RPS.

Question Source: Bank # _____
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A_____

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Control Rod Drive	Group	2		
	K/A	001 A2.03		
	Importance Rating	3.5		
Ability to a) predict the impacts of the following malfunctions or operations on the CRDS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effect of stuck rod or Misaligned rod				

Question # 56

Reactor Power was 100% when a secondary load rejection occurs.

- Power is reduced to 81% before stabilizing
- Control rods (CR) M-4 and M-12 in bank D failed to move with the rest of the bank
- The following annunciators are received:
  - 78A, PR CHANNEL DEV
  - 79C, CONTROL ROD DEV
  - 80C, RPI ROD DEV

Control Bank D is at 195 steps and Rods M-4 and M-12 are at 215 steps.

(1) What describes the INITIAL actions required?

And

(2) The reason for the time limit requirement for realignment of rods M-4 and M-12?

A. (1) Reduce Thermal Power to < 50% within 30 minutes
(2) The core is not analyzed for more than one misaligned control rod

B. (1) Reduce Thermal Power to < 50% within 30 minutes
(2) Local xenon redistribution may potentially cause excessive power peaking.

C. (1) Verify Shutdown Margin is within limits within 1 hour of the failure of Control rods M-4 and M-12 to move
(2) The core is not analyzed for more than one misaligned control rod

D. (1) Verify Shutdown Margin is within limits within 1 hour of the failure of Control Rods M-4 and M-12 to move

(2) Local xenon redistribution may potentially cause excessive power peaking

Answer: D

Explanation:
A: INCORRECT:RNO for step A5 of OTO-SF-00001 is not correct and core not analyzed is not correct
B: INCORRECT. RNO for step A5 of OTO-SF-00001 is not correct even though Xenon redistribution is correct
C: INCORRECT: Step A4 of OTO-SF-00001is correct but core not analyzed is not correct
D: CORRECT: Step A4 of OTO-SF-00001 is correct and Xenon redistribution is correct

Technical Reference(s):
1. OTO-SF-00001, Rod Control Malfunctions, Rev 17

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP #45, Objective D: Given a set of plant conditions or parameters indicating a Rod Control Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # _____
Modified Bank # _16522
New _______

Question History: Last NRC Exam __N/A__

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Reactor Coolant	**Group**	2		
	K/A	002 A4.01		
	Importance Rating	3.5		
Ability to manually operate and/or monitor in the control room: RCS leakage calculation program using the computer				

Question # 57

The crew is responding to an RCS leak per OTO-BB-00003, Excessive RCS Leakage.

The following PPC data has been gathered:

	Time = 0	Time = 1 minute
- Reactor Power (%)	100	100
- RCS Tave (°F)	585.3	585.3
- Charging Flow (gpm)	140	140
- Letdown Flow (gpm)	75	75
- Pzr Level (%)	56	55.7
- Total Seal Injection Flow (gpm)	32	32
- Total Seal Leakoff Flow (gpm)	12	12

The PPC should be calculating a RCS Leak Rate of ……

A. 18 gpm

B. 35 gpm

C. 53 gpm

D. 71 gpm

Answer: D

Explanation:

A: INCORRECT: 18 gpm is the change of PZR level only and doesnt account for the gharging / letdown mismatches.
B: INCORRECT 140 - 87 gpm -18 gpm = 35 gpm distractor.
C: INCORRECT: 140 - 87 gpm = 53 gpm distractor
D: CORRECT 140 - 87 gpm but then the change in PZR level must be accounted for. Using the thumbrule of 60 gallons = 1% while hot means that 18 gallons PZR was lost in that minute. Therefore 18 gpm must be added to the 53 gpm for the correct answer of 71 gpm.

Technical Reference(s):
1. OSP-BB-00009, Addendum 1, RCS Inventory Balance Excessive Leakage or Manual Calculation, Rev 5

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP#12, Objective F: Using the plant computer or data provided by the instructor, DETERMINE the amount of excessive RCS leakage. CONFIRM the leakage rate using charging and letdown mismatch and CLASSIFY the leak rate in terms of the required Plant Shutdown and Cooldown.

Question Source: Bank # _X___R23343__
Modified Bank # ______
New ______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Pressurizer Level Control	Group #	2		
	K/A #	011 K1.01		
	Importance Rating	3.6		
Knowledge of the physical connections and/or cause-effect relationships between the PZR LCS and the following systems CVCS				

Question # 58

Reactor Power is 6%. The Pressurizer level control systems are in automatic, with the NCP in service.

BB-LT-461, the UPPER selected Pressurizer level channel, fails LOW.

(1) If no operator action is taken, what is the response of BGFCV0124, NCP Flow Control Valve?

And

(2) What are the effects on the plant?

A. (1) It remains the same
(2) Letdown isolates, and pressurizer heaters turn OFF ONLY

B. (1) It remains the same
(2) Letdown isolates, pressurizer heaters turn OFF, and the Reactor trips on High PZR water level

C. (1) It throttles open
(2) Letdown isolates, and pressurizer heaters turn OFF ONLY

D. (1) It throttles open
(2) Letdown isolates, pressurizer heaters turn OFF, and the Reactor trips on High PZR water level

Answer: C

Explanation:
Per OTO-BG-00001, Attachment A for LT-461 when it is the LOWER selected channel, the effects are as follows:

- *Turns off all Pzrs and isolates letdown on Pzr level (17% LCV460 and Thrtl Iso vlvs close).*

- *Provides low level alarm (Ann 328 Pzr 17% HTRS OFF LTDN ISO)*
- *Provides high level alarm (70%-Ann 32A Pzr Lev HI).*

If it was selected for UPPER control and Alarm:

- *Turns off all Pzr heaters and isolates letdown on low Pzr level. (17% LCV459 and Thrtl Iso vlvs close)*
- *Provides low level alarm. (Ann 328 Pzr 17% HTRS OFF LTDN ISO).*
- *Provides low level dev alarm 5% below setpoint (Ann 32C Pzr LO LVL DEV)*
- *Provides high level dev alarm 5% above setpoint (Pzr HI LEV DEV HTRS ON) and energizes Pzr Heaters.*
- *Provides charging flow demand signal to FCV- and 124 when in auto.*

Therefore the lower selected channel provides no input into the NCP FCV control but the UPPER channel will provide an input into its controller/positioner.

As the Reactor is below P-7 and per E-0 Section B: the PZR water level - High (2/3, P-7) is NOT enabled will not cause a reactor trip but it is plausible as letdown has isolated and NCP flow will be going up. P-7 is either P-10 or P-13 (which are both 10% NI or turbine Imp pressure)

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Correct – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):
1. OTO-BG-00001, Pressurizer Level Control Malfunctions, Rev 23
2. E-0, Reactor Trip or Safety Injection, Rev 20 Section B

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP #11 Objective B: DESCRIBE the purpose, operation and interlocks for the following CVCS components:

1. Letdown Isolation Valves
2. Delay Pipe
3. Regenerative Heat Exchanger (Regen Hx)
4. Letdown Throttle Isolation Valves
5. Letdown Throttle Valves
6. Letdown Containment Isolation Valves
7. Letdown Reheat Divert Valve (BGTCV0381B)
8. Letdown Hx
9. Letdown Hx Outlet Pressure Control Valve (BGPCV0130)
10. CVCS Demineralizer Inlet Divert Valve (BGTCV0129)
11. CVCS Demineralizers
12. Reactor Coolant Filter
13. Volume Control Tank (VCT) Inlet Divert Valve (BGLCV0112A)
14. VCT
15. VCT Outlet Valves (BGLCV0112B and BGLCV0112C)
16. Charging Pump Suction Valves
17. Normal Charging Pump (NCP)
18. Centrifugal Charging Pumps (CCPs)

19. CCP/NCP Flow Control Valves (BGFCV0121 and BGFCV0124)

T61.0110, Systems, LP #30, Rx Instrumentation, Objective A: STATE the functions and EXPLAIN the design criteria of the Pressurizer Pressure/Level Control Systems.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	2		
017 – In-core Temperature Monitor	**Group #**	2		
	K/A #	017 A1.01		
	Importance Rating	3.7		
Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM system controls including: Core exit thermocouples.				

Question # 59

The crew has entered E-1, Loss of Reactor or Secondary Coolant.

Current plant conditions are as follows:

- Power Range Indications (All four) ~3%
- Intermediate range startup rate +0.2 dpm
- RCPs none running
- Core Exit TCs 1220°F
- RCS subcooling 20°F
- RVLIS PUMPS OFF indication 31%
- Containment Pressure 28 psig
- Containment Spray Pumps none running
- All S/G narrow range levels ~6%
- Total AFW Flow 200,000 lbm/hr

What procedure should the operators enter FIRST?

A. FR-S.1, Response to Nuclear Power Generation/ATWS

B. FR-C.1, Response to Inadequate Core Cooling

C. FR-Z.1, Response to High Containment Pressure

D. FR-H.1, Response to Loss of Secondary Heat Sink

Answer: B

Explanation:

A. Incorrect. FR-S.1 has priority than FR-C.1 but it is only an orange path
B. Correct. Conditions meet FR-C.1 and is a red pad
C. Incorrect. FR-Z.1 is an orange path
D. Incorrect. FR-H.1 is a red path but lower priority than FR-C.1

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Technical Reference(s):
1. CSF-1, Critical Safety Function Status Trees (CSFST), Rev 013

References to be provided to applicants during examination: None

Learning Objective: T61.003D 6, D-01 ERG Introduction & User's Guide, Objective K. Explain how challenges to critical safety functions are prioritized within each critical safety function..

Question Source: Bank # ___
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Hydrogen Recombiner and Purge Control	Group #	2		
	K/A #	028 K5.01		
	Importance Rating	3.4		
Knowledge of the operational implications of the following concepts as they apply to the HRPS: Explosive hydrogen concentration.				

Question # 60

The plant experienced a LOCA three hours ago.

You have been directed to place an Electric Thermal Hydrogen Recombiner (ETHR) in service.

What is the LOWEST Containment Hydrogen volumetric concentration that would PREVENT the ETHR from being placed in service?

A. 3%

B. 4%

C. 5%

D. 6%

Answer: B

Explanation:
In accordance with OTN-GS-00001, step 5.5.1 “IF containment hydrogen volumetric concentration is greater than or equal to 4%, do NOT place an ETHR in service.” Therefore, 4% is correct as it is the LOWEST that would prevent the ETHR from being placed in service per the procedure.

A. Incorrect – See above Explanation
B. Correct – See above Explanation,
C. Incorrect – See above Explanation
D. Incorrect – See above Explanation

Technical Reference(s):
1. OTN-GS-00001, Containment Hydrogen Control System, Rev 17

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References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP-40 Q. EXPLAIN the precautions, limitations and bases for the following processes/conditions associated with OTN-GS-00001, "Containment Hydrogen Control System":

01. Hydrogen concentration with Hydrogen Recombiner in service or being placed in service.

Question Source: Bank # __X____
Modified Bank # ______
New ______

Question History: Last NRC Exam 2014 Q#58_

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(5)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Spent Fuel Pool Cooling	Group #	2		
	K/A #	033 K3.03		
	Importance Rating	3.0		
Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: Spent fuel temperature				

Question # 61

Spent fuel handling operations in the Spent Fuel Pool (SFP) are in progress.

The running spent fuel pool cooling pump tripped and the standby pump could not be started.

The crew has just entered OTO-EC-00002, Spent Fuel Pool High Temperature.

(1) Per OTO-EC-00002, what is the MAXIMUM SFP Temperature allowed before All Fuel Handling operations must be stopped?

And

(2) If after all fuel handling is secured and temperature continues to rise with no SFP Cooling pump running, what is the NEXT action the crew should take per OTO-EC-00002?

A. (1) 113°F
(2) Bypass SFP Demineralizers

B. (1) 113°F
(2) Initiate a Fuel Building Isolation Signal (FBIS)

C. (1) 140°F
(2) Bypass SFP Demineralizers

D. (1) 140°F
(2) Initiate a Fuel Building Isolation Signal (FBIS)

Answer: A

Explanation:

OTO-EC-00002 Step 3 RNO "Stop all fuel handling operations" occurs when the limit of the Curve book has been exceeded. This limit is 113F. The Distractor of 200F is from continuous action step #8 where the RNO directs the radwaste operator to bypass SFP Demins.

The distractor of initiate FBIS is plausible as it may be believe the actions are to bottle up the Fuel Building to prevent or contain any potential releases from the SFP. This action is not in OTO-EC-00002. Step #8 RNO where the radwaste operator is directed to isolate the Demin bed

A. Correct
B. Incorrect – see above explanation
C. Incorrect – see above explanation
D. Incorrect – see above explanation

Technical Reference(s):
1. Curve Book Table 8-8b
2. OTO-EC-00002, Spent Fuel Pool High Temperature, Rev 11
3. OTA-RK-00022, ADD 75D, SFP Temp Hi, Rev 3

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #24, Objective I: EXPLAIN the precautions, limitations and bases for the following conditions/processes associated with OTN-EC-00001, "Fuel Pool Cooling and Cleanup System":
1. Minimum Spent Fuel Pool (SFP) level
2. SFP temperature limits
3. Minimum boron concentration
4. Requirements for placing cleanup in service
5. Skimmer operation restrictions when cleanup is aligned to the RWST

Question Source: Bank # ______
Modified Bank # ______
New __X_____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(4)

Comments:
K/A match as the knowledge and/or effect on the loss of a SFP cooling was examined through knowledge of SFP temperature limits and procedurally required actions as the temperature goes up making it operationally valid to control room operators.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Steam Generator	Group	2		
	K/A	035 A3.01		
	Importance Rating	4.0		
Ability to monitor automatic operation of the S/G including: S/G water				

Question # 62

Reactor Power is 100% when one of the level channels for 'C' Steam Generator fails to 0%.

Assuming no operator action, feedwater flow to ‘C’ Steam Generator will…

A. rise until a P-14 actuation occurs

B. lower until the reactor trips on Lo-Lo Steam Generator Level

C. initially rise, then will lower resulting in a steam generator level stabilizing higher than before

D. initially lower, then will rise resulting in a steam generator level stabilizing lower than before

Answer: A

Explanation:

A: CORRECT: A level error output signal is produced that now causes the MFRV to open. The increased flow produces a flow error, but it cannot overcome the level error output signal. Therefore the level continues to increase until a Feedwater Isolation Actuation occurs (2/4 S/G levels > 91%).
B: INCORRECT: Steam Generator Level increases in this failure, not decreases.
C: INCORRECT: FW flow will increase as indicated, but will not go back down because the controlling channel is failed low and will not change with actual level.
D: INCORRECT: When one of the channel fails to 0% a level error is created that indicates the actual level is too low. This causes the MFRV to open, increasing FW flow, not decreasing it as indicated.

Technical Reference(s):
1. M-22AE02, P&ID Feedwater System, Rev 32
2. OTO-AE-00002, SG Water Level Control Malfunction, Rev 13

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off normal Operations, LP#40, Objective D: Given a set of plant conditions or parameters indicating a Steam Generator Water Level Control Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ___
Modified Bank # ______
New ___X___

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	2		
Steam Dump/Turbine Bypass Control	Group #	2		
	K/A #	041 K6.03		
	Importance Rating	2.7		
Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS				

Question # 63

Reactor Power is 100%.

Unknown to the operators, one Tavg channel output is failed 'as is' at 585.25°F.

If no operator action is taken following a reactor trip, the Condenser Steam Dump valves will …….

A. cool the RCS to 557°F and then throttle to maintain this temperature

B. cool the RCS to 550°F and then cycle closed and open at this temperature

C. open until Main Steam pressure reaches 1092 psig and then throttle to maintain this pressure

D. open until Main Steam pressure reaches 615 psig and then MSIVs close and the RCS begins to heat up again

Answer: B

Explanation:

When the reactor trips and the steam dumps transfer from load reject mode to plant trip mode, they will be armed and see a Tref vs Tavg mismatch of 557F to 585.25F. This will cause the steam dumps to open and remain when until 550F (SD control system will deenergize a solenoid at 550F which will close the SDs).

A. Incorrect – this is the proper response when controlling RCS Tavg after a reactor trip with no complications
B. Correct
C. Incorrect – When the steam dump controller is in steam pressure mode, the controller maintains steam header pressure of 1092 which translate to a no load RCS Tavg of 557F
D. Incorrect but plausible if the candidate believes that the open steam dumps will lower main steam header pressure to 615 psig post trip. Saturated temperature at 615 psig is 492F, therefore while the process of a cooldown is correct, this will NOT occur as the steam dumps will be closed

when RCS temp reaches 550F.

Technical Reference(s):
1. OTO-AB-00001, Steam Dump Malfunction, Rev 18
2. M-22AB03, P&ID Main Steam System, Rev 27
3. OTN-AB-00001, Main Steam and Steam Dump Systems, Attachment 4, Rev 23

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #20, Objective D: IDENTIFY all Main Steam, Steam Dump and S/G controls, alarms and indications and DESCRIBE how each is used to predict, monitor or control the Main Steam, Steam Dump and S/G System.

Question Source: Bank # ___X R12183 and 2017 ILT Audit Q62___
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	2		
Condensate	Group	2		
	K/A	056 G2.1.20		
	Importance Rating	4.6		
Ability to interpret and execute procedure steps				

Question # 64

Reactor Power is 17%.

- A & C Condensate Pumps in-service
- B Main Feed Pump in-service
- PB03 bus Lockout occurs

(1) What is the FIRST automatic action to occur due to the loss of PB03?

And

(2) What action would be performed NEXT?

A. (1) MD AFAS
(2) shutdown reactor per OTG-ZZ-00005

B. (1) MD AFAS
(2) trip reactor per OTO-AE-00001

C. (1) TD AFAS
(2) shutdown reactor per OTG-ZZ-00005

D. (1) TD AFAS
(2) trip reactor per OTO-AE-00001

Answer: B

Explanation:
-PB03 lost power causing A and C Condensate pumps to trip, this caused the B MFP to trip and AFAS-M

A – Incorrect. OTA-RK-00026 Add123A, MFP B TRIP, step 3.4 provides actions when the MFP trips to shutdown reactor per OTG-ZZ-00005 if Aux Feedwater is only source of water and main generator not synched to grid.
B – Correct – Per OTA-RK-00026 Add123A, MFP B TRIP, go to OTO-AE-00001 if Generator synched to grid and per step 1 trip rx if all MFPs tripped
C – Incorrect –criteria for TD AFAS is not met

D – Incorrect –criteria for TD AFAS is not met

Technical Reference(s):
1. OTA-RK-00024 Addendum 123A, MFP B TRIP
2. OTO-AE-00001, Feedwater System Malfunction

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #22, Objective G: IDENTIFY the Condensate and Condenser Air Removal System Main Control Board (MCB) controls, alarms and indications and DESCRIBE how each is used to predict, monitor and control changes in the systems.

Question Source: Bank # _____
Modified Bank # _8524_
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	2		
Circulating Water	**Group #**	2		
	K/A #	075 K2.03		
	Importance Rating	2.6		
Knowledge of bus power supplies to the following: Emergency/essential SWS pumps				

Question # 65

What is the power supply for the 'B' ESW Pump, DPEF01B?

A. NG05

B. NG06

C. NB01

D. NB02

Answer: D

Explanation:

Per the reference, the power supply to the B ESW pump is NB02 via breaker NB0215. The distractor of NG06 is plausible as it is a safety related pump house (where the ESW pump is physically related) MCC. The distractor of NB01 is plausible as it is 4160 VAC and safety related but incorrect as it power the other safety train. The distractor of NG05 is also plausible is it is located at the pump house but is wrong as it is 480 VAC and power from the other safety train.

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Incorrect – See above explanation
D. Correct – See above explanation

Technical Reference(s):
1. E-21NB02(Q), Lower Medium Voltage System Class 1E, Rev 17

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #5, Objective B: DESCRIBE the purpose and operation of the following Essential Service Water System components:
1. Essential Service Water Pump

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Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(7)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	3		
Conduct of Operations	Group	Generic		
	K/A	G2.1.1		
	Importance Rating	3.8		
Knowledge of Conduct of operations requirements				

Question # 66

The operating shift crew compliment may be _(A)__less for a period of time not to exceed _(B)__hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift compliment.

	(A)	(B)
A.	1	2
B.	1	1
C.	2	1
D.	2	2

Answer: A

Explanation:

A is correct as stated in step 4.3.1.C of ODP-ZZ-00001
B in incorrect due to hours
C is incorrect due to number of persons absents and hours
D is incorrect due to persons absent

Technical Reference(s):
1. ODP-ZZ-00001 step 4.3.1.C, Operation Department Code of Conduct, Rev 104

References to be provided to applicants during examination: None

Learning Objective: T61.003A, Normal Operations, LP #28, Objective A: EXPLAIN the following as applied in ODP ZZ 00001, Operations Dept. – Code of Conduct:

7. DISCUSS:
   a. Minimum Shift Manning requirements
   b. Unexpected absence requirements regarding shift complement

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Question Source: Bank # _X___L5966__
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A_____

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier	3		
Conduct of Operations	**Group**	Generic		
	K/A	G2.1.20		
	Importance Rating	4.6		
Ability to interpret and execute procedure steps				

Question # 67

Per ODP-ZZ-00025, EOP/OTO User's Guide what correctly describes when an emergency procedure action on the Foldout Page is applicable?

A. Only PRIOR to performing the applicable step in the main body of the procedure

B. ANY time during the applicable procedure performance, unless a specific procedural starting point is referenced in the action

C. Only after proceeding PAST the applicable step in the main body of the procedure, AND it MAY apply after a transition is made to another procedure

D. Only after proceeding PAST the applicable step in the main body of the procedure, BUT it DOES NOT apply after a transition is made to another procedure

Answer: B

Explanation:

A Incorrect, FOP actions apply as soon as the procedure is entered.
B Correct ODP-ZZ-00025 states that a step on the Foldout Page contains information that must be monitored throughout the procedure.
C Incorrect, however, this may be applicable to continuous action summary steps depending on the action and whether it is superseded or no longer applicable to the next procedure.
D Incorrect, however, this may be applicable to continuous action summary steps depending on the action and whether it is superseded or no longer applicable to the next procedure.

Technical Reference(s):
1. ODP-ZZ-00025, EOP/OTO User's Guide, Rev 34

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP#1, Objective H: DISCUSS

procedural requirements for ODP-ZZ-00025, EOP/OTO User's Guide, to include:

- Command and Control
- Step Sequencing

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Conduct of Operations	Group #	Generic		
	K/A #	G2.1.37		
	Importance Rating	4.3		
Knowledge of procedures, guidelines, or limitations associated with reactivity management.				

Question # 68

Per APA-ZZ-01300, Reactivity Management Program, which of the following meets the criteria for a Significance Level 1 Reactivity Event?

A. The RO over dilutes and allows inadvertent criticality of reactor.

B. The RO takes the reactor critical 450 pcm above Estimated Critical Position.

C. The RO violates procedure and exceeds the steady state SUR of 1.0 DPM.

D. The RO misadjusts PR NIs such that indicated power is 2% higher than actual power.

Answer: A

Explanation:

Per APA-ZZ-01300 Attachment 1 the following are Examples of SL 1 Events:

1.0 Other issues meeting the SL 1 criteria
1.1 Control Rod ejection
1.2 Anticipated transient without trip
1.3 Violation of a Reactor Core Safety limit
1.4 Improper reactivity control that results in fuel failures greater than Technical Specification (T/S) limits
1.5 Reactor criticality with misconfigured, misoriented or mislocated fuel or control component in the core
1.6 Inadvertent criticality:
1.7 SL 2 or 3 event resulting from a fundamental organizational breakdown

A. Correct – See above
B. Incorrect – this is an example of a safety level 2 event " Criticality occurs outside the predetermined acceptance criteria"
C. Incorrect – this is an example of a safety level 3 event " Violation of Core Thermal Power procedural (administrative) limit"

D. Incorrect – this is an example of a safety level 4 event " Unexplained (after evaluation) discrepancies between alternate indications of reactor power (not meant to include instrument failures)"

Technical Reference(s):
1. APA-ZZ-01300, Reactivity Management Program, Rev 28

References to be provided to applicants during examination: None

Learning Objective: T61.003A, Normal Operations, LP #1, Objective A: SPECIFY the following as it pertains to APA-ZZ-01300, Reactivity Management Program:

1. The purpose and scope
2. The definitions of:
    a. Reactivity
    b. Reactivity Control
    c. Reactivity Event
3. Responsibilities of:
    a. Licensed Operators
    b. Reactor Engineering
    c. Work Management
4. Reactivity Management Philosophy
5. RECOGNIZE and provide examples of events for different significance levels.
6. DISCUSS SOER 07-1, Reactivity Management

Question Source: Bank # _______
Modified Bank # _13135__
New _______

Question History: Last NRC Exam __N/A____

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Equipment Control	**Group #**	Generic		
	K/A #	G2.2.6		
	Importance Rating	3.0		
Knowledge of the process for making changes to procedures.				

Question # 69

A Temporary Change can be used to change which of the following?

A. Operator Aid

B. Work Instructions

C. Vendor Procedure

D. Normal Operating Procedure

Answer: D

Explanation:
Per APA-ZZ-00101, step #4.4.3.a states “ Initiator Temporary Change”
a. CHECK Temporary Change is NOT against the following:

- *Emergency Operating Procedures*
- *Off-Normal Procedures*
- *Severe Accident Management Guidelines*
- *Forms*
- *Procedure-Based Planning*
- *Computer-based Checklists and Checkoff Lists*
- *Vendor Procedures*
- *Manuals*
- *Operator Aids*
- *Work Instructions*
- *Electronic Procedures as defined in Step 7.23*

And if any of the above is true then the change would be a minor or major revision per step #4.4.4.

A. Incorrect – see above.
B. Incorrect – see above.
C. Incorrect – see above.
D. Correct – see above.

Technical Reference(s):
1. APA-ZZ-00101, Processing Procedures, Manuals, and Desktop Instructions, Rev 75.

References to be provided to applicants during examination: None

Learning Objective: T61.003A, Normal Operations, LP #14, Objective F: STATE the following as they pertain to APA-ZZ-00101 – Processing Procedures, Manuals, and Desktop Instructions:

- The Purpose and Scope
- When Administrative Correction Revisions may be performed
- When Temporary Changes may be performed
- SRO role in Temporary Change process
- Reviews required for Major/Minor Revisions and New Procedures

Question Source: Bank # __ X – no bank id – 2016 Audit Exam Q#68_
Modified Bank # ______
New _______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Equipment Control	Group #	Generic		
	K/A #	G2.2.13		
	Importance Rating	4.1		
Knowledge of tagging and clearance procedures.				

Question # 70

An Operations Technician is in the process of performing an Independent Verification (IV) of Workman's Protection Assurance (WPA) placed in the plant.

Per ODP-ZZ-00310, WPA, Local Control and Caution Tagging, which of the following radiation levels would permit the Independent Verification (IV) to be waived?

A. Airborne activity in the area is 1.5 DAC

B. General Area Dose Rate is 15 mRem/Hr

C. Contamination in the area is 15 dpm/100 cm^2

D. An Exposure of 15 mRem is likely to be received

Answer: D

Explanation: *Per the technical reference, (Note on page 26 that states 'The waiver of an IV is allowed when General Area Dose Rates are greater than 25mRem/Hr, or in situations where radiation exposures of greater than 10 mRem are likely) the waiver of an IV is allowed when radiation exposures is expected to be greater than 10mR.*

A. Incorrect. There is no specific limit on airborne activity.
B. Incorrect. The value for general dose rates is 25mR/hr.
C. Incorrect. There is no specific limit on area contamination but plausible as contamination leads to dose and preventing the spread of contamination (via access and movement in the area) could spread contamination which should be prevented.
D. Correct

Technical Reference(s):
1. ODP-ZZ-00310, WPA, Local Control and Caution Tagging, Rev 78

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Learning Objective: T61.003A, Normal Operations, LP #13, WPA.

A. ODP-ZZ-00310, WPA and Caution Tagging

1. **DISCUSS** the Responsibilities of the following:

a. Shift Manager/Control Room Supervisor

1. Discus when an Independent Verification may be waived.

Question Source: Bank # __X R6464____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2014 ILT NRC Exam Q#70________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier #	3		
Radiation Control	**Group #**	Generic		
	K/A #	G2.3.5		
	Importance Rating	2.9		
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.				

Question # 71

While taking rounds the RO notices the RM-11 has a channel indicating “Dark Blue”.

What does this indicate to the operator?

A. Check Source is in progress.

B. The monitor has been taken out of poll.

C. A system failure associated with the channel has occurred.

D. A communications failure between the RM11 and the monitor (RM80) has occurred.

Answer: C

Explanation: *See Attachment 2 of OTN-SP-00002*

A. Incorrect. This would yield a “Gray” indication IAW OTN-SP-00002.
B. Incorrect. This would yield a “White” indication IAW OTN-SP-00002.
C. Correct. This is correct IAW OTN-SP-00002.
D. Incorrect. This would yield a “Magenta” indication IAW OTN-SP-00002..

Technical Reference(s):
1. OTN-SP-00002, Radiation Monitor Control Panel RM-11, Rev 8

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP#36, Obj C .IDENTIFY the Process and Area Radiation Monitoring Control Room controls, alarms, and indications and DESCRIBE how each is used to predict, monitor and control the Process and Area Radiation Monitoring System.

Question Source: Bank # __ 2014 Audit Exam Q#71____
Modified Bank # ______

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(11)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Radiation Control	Group #	Generic		
	K/A #	G2.3.7		
	Importance Rating	3.5		
Ability to comply with radiation work permit requirements during normal or abnormal conditions.				

Question # 72

Operators are preparing to isolate a 2 gpm leak in the Auxiliary Building.

Per the RWP, radiological survey information is as follows:

- Dose Rate is 600 mrem/hr @30cm
- Rate Alarm setpoint is 900 mrem/hr
- Dose Alarm is 1800 mrem
- 80,000 dpm/100cm^2 surface contamination

The job is expected to take 2 operators.

What should the general area be posted as and when would the operator receive a dosimeter alarm?

A. High Radiation Area; 1.5 hrs

B. High Radiation Area; 3.0 hrs

C. Locked High Radiation Area; 1.5 hrs

D. Locked High Radiation Area; 3.0 hrs

Answer: B

Explanation:

Per HDP-ZZ-01500, a locked HRA is "A Locked High Radiation Area is any area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent greater than 1000 mrem/hour at 30 centimeters, but less than 500 rads/hour at 1 meter from radiation source or from any surface that the radiation penetrates."

And a High Radiation Area is "A High Radiation Area (HRA) is any area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 100 mrem/hour and NOT exceeding 1000 mrem / hour at 30 centimeters from radiation source or from any surface that the radiation penetrates."

Locked high radiation area is plausible if the candidate falsely uses the dose alarm setpoint of 1800mrem instead of the actual 600 mrem/hr

Stay time is calculated per individual not per group. With the numbers given in the stem, an individual could stay in the dose field for 3 hours (1800 mrem / 600mrem per hour) = 3 hours.

The Distractor of 1.5 hours is if the candidate believes that stay time is calculated per group dose i.e there are 2 operators so each operator can only receive ½ of the dose alarm (0.5 x 1800 = 900 mrem) and the 900 mrem / 600 mrem per hour is 1.5 hours. Additionally, the candidate may falsely use the incorrect number for dose rate alarm even if the understand stay time is calculated per person. I.e if they use 900 mrem/ hr / 600mrem/hr and arrive at 1.5. Thus 1.5 hours is a plausible distractor.

A. Incorrect – See above explanation,
B. Correct – See above explanation
C. Incorrect – See above explanation,
D. Incorrect – See above explanation,

Technical Reference(s):
1. HDP-ZZ-01500, Radiological Posting, Rev 48

References to be provided to applicants during examination: None

Learning Objective: T61.003A, Normal Operations, LP #33, Objective A: DISCUSS the following as they apply to HDP-ZZ-06100, "Reactor Building Access":
1. Purpose and scope
2. Precautions and limitations (emergency and non-emergency)
3. Mode 5 and 6 Equipment Hatch opening requirements.
4. Discuss the process for a normal containment entry.
5. Discuss the process for an emergency containment entry.

Question Source: Bank # __X – no bank id____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2017 NRC ILT Exam Q#72________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(12)

Comments:

2 gpm leak is recent Plant OE from boric acid corrosion and leakage past a flange.

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier	3		
Emergency Procedures / Plan	**Group**	Generic		
	K/A	G2.4.1		
	Importance Rating	4.6		
Knowledge of EOP entry conditions and immediate action steps.				

Question # 73

The plant loses power to NB01 and NB02 due to bus lockouts.

(1) What immediate action procedural steps are required for this event?

And

(2) Per ODP-ZZ-00025, EOP/OTO User's Guide, Immediate Actions steps are required to be performed _____________?

A. (1) Complete steps 1 through 4 of E-0
(2) from memory

B. (1) Complete steps 1 through 2 of ECA-0.0
(2) from memory

C. (1) Complete steps 1 through 4 of E-0
(2) using procedure

D. (1) Complete steps 1 through 2 of ECA-0.0
(2) using procedure

Answer: B

Explanation:
ODP-ZZ-00025
Step 4.2. Entry Into EOP Network
Entry into the EOP Network is limited to three specific conditions:
• If the plant is in Modes 1, 2, or 3 and the Safety Injection accumulator outlet valves are open, E-0, Reactor Trip Or Safety Injection is entered if a reactor trip or safety injection occurs or is required.
• If the Plant is in Modes 1 through 4, a complete loss of power on the AC emergency buses occurs and one train cannot be restored quickly, the operator should enter ECA-0.0, Loss Of All AC Power.
• If the Plant is in Mode 5 or 6, a complete loss of power occurs, the Operator should enter OTO-NB-00005, Loss of All AC Power While on RHR
Step 4.7.2. Immediate Actions steps are required to be performed from memory.

ECA-0.0 Note; Steps 1 and 2 are immediate action steps..

A:Incorrect - E-0 would not be entered since Rx is not tripped and if Rx tripped, the crew would exit during step 3 of E-0 and not complete step 4
B Correct- See above
C:Incorrect – E-0 would not be entered since Rx is not tripped and if Rx tripped, the crew would exit during step 3 of E-0 and not complete step 4 and Immediate action steps are from memory
D:Incorrect -Immediate action steps are from memory

Technical Reference(s):
1. ODP-ZZ-00025 EOP/OTO USER'S GUIDE, Rev 34
2. ECA-0.0, Loss Of All AC Power, Rev 029

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP#1, Objective H DISCUSS procedural requirements for ODP-ZZ-00025, EOP/OTO User's Guide, to include:
- Command and Control
- Step Sequencing
- Immediate Actions

Question Source: Bank # ______
Modified Bank # ______
New ___X______

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.41(10)

Comments:

Examination Outline Cross-reference:	Level	RO		Rev 0
	Tier #	3		
Emergency Procedures / Plan	Group #	Generic		
	K/A #	G2.4.13		
	Importance Rating	4.0		
Knowledge of crew roles and responsibilities during EOP usage.				

Question # 74

An event has occurred and the operating crew has entered the EOPs.

(1) Who is responsible to provide a peer check to the Control Room Supervisor to ensure the correct diagnosis, procedure flowpath and transitions?

And

(2) Per ODP-ZZ-00025, EOP/OTO User's Guide, during the concurrent performance of an EOP and OTO, a procedure conflict is identified. The ________________ is responsible to choose the action that is most appropriate for the existing plant conditions.

A. (1) Shift Manager
(2) Shift Manager

B. (1) Shift Manager
(2) Control Room Supervisor

C. (1) Incident Assessor
(2) Shift Manager

D. (1) Incident Assessor
(2) Control Room Supervisor

Answer: D

Explanation:
IAW ODP-ZZ-000025 step 3.6.3 the incident Assessor "Provides a peer check to the CRS to ensure correct diagnosis, procedure flowpath and transitions." Furthermore, step 4.27.2 identifies that it is the CRS responsibility to determine the action that is most appropriate for the existing plant conditions. "Procedure Conflict - If procedures are implemented concurrently, preplanned actions in the procedures may conflict due to the assumptions used in the design of the procedures. When a conflict exists, it is the responsibility of the CRS to choose the action(s) that is most appropriate for the existing plant conditions."

A. Incorrect – both are wrong
B. Incorrect – the IA is responsible not the shift manager
C. Incorrect – The CRS is responsible, not the shift manager
D. Correct

Technical Reference(s):.
1. ODP-ZZ-00025, Rev 34, Section 3.6 and 4.27

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal, LP #1, Intro to Off normal Procedures, Objective F: DISCUSS the responsibilities addressed in ODP-ZZ-00025, EOP/OTO User's Guide, for the following positions.

- Shift Manager (SM)
- Control Room Supervisor (CRS)
- Incident Assessor (IA)
- Reactor Operator (RO)
- Operations Technician (OT)

Question Source: Bank # _X_ no bank id 2016 Audit Exam Q#74
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	**Level**	**RO**		**Rev 0**
	Tier	3		
Emergency Procedure / Plan	**Group**	Generic		
	K/A	G2.4.27		
	Importance Rating	3.4		
Knowledge of "fire in the plant" procedures				

Question # 75

The fire brigade has been activated for a major fire located in the Stores Building.

Who should be contacted FIRST if additional fire brigade manning is needed after the initial announcement?

A. Fire Protection Engineer

B. Engineering Duty Supervisor (EDS)

C. Offsite Qualified fire brigade members

D. Fire Brigade Personnel that are in Training

Answer: D

Explanation:

A: INCORRECT: Step 5.2.4.h of EIP-ZZ-00226
B: INCORRECT : Step 5.2.4.g of EIP-ZZ-00226
C: INCORRECT Step 5.3 of EIP-ZZ-00226
D: CORRECT Step 5.2.3.e of EIP-ZZ-00226

Technical Reference(s):
1. EIP-ZZ-00226, Fire Response Procedure for Callaway Plant, Rev 20

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off normal Operations, LP #51 Objective A: DESCRIBE the following as they pertain to EIP-ZZ-00226, "Fire Response Procedure for Callaway Plant":
1. Purpose and scope
2. Shift Manager responsibilities
3. Control Room personnel responsibilities/actions upon report of a fire

Question Source: Bank # __6136__

NRC Site-Specific Written Examination
Callaway Plant
Reactor Operator

Modified Bank # ______
New ______

Question History: Last NRC Exam ____N/A_____

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

10 CFR Part 55 Content:

10 CFR 55.41(b)(10)

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	1		
Pressurizer Vapor Space Accident	Group	1		
	K/A	008 AA2.26		
	Importance Rating	3.4		
Ability to determine or interpret the following as they apply to PZR vapor space accident: Probable PZR steam space leakage paths other than PORV or code safety				

Question # 76

An automatic reactor trip and safety injection has occurred as a result of lowering RCS pressure.

The crew has just entered E-1, Loss of Reactor or Secondary Coolant and plant conditions are as follows:

	Prior to SI	After SI Occurs
Pzr Pressure	Lowering	Lowering
RCS Temperature	Stable	Stable
Pzr Level	Stable	Rising

(1) What accident would result in these conditions?

And

(2) The CRS will direct what procedure NEXT?

A. (1) RCS Cold Leg Leak
(2) ES-1.1, SI Termination

B. (1) RCS Cold Leg Leak
(2) ES-1.2, Post LOCA Cooldown and Depressurization

C. (1) PZR Vapor Space Leak
(2) ES-1.1, SI Termination

D. (1) PZR Vapor Space Leak
(2) ES-1.2, Post LOCA Cooldown and Depressurization

Answer: D

Explanation:

Based on the fact that RCS temperature is stable, it is not a RCS cold leg leak. With RCS Temperature stable and RCS Pressure lowering and RCS level rising, it is a PZR steam space leak (in the case a PZR Spray line)

Based on the conditions present after SI and per step #6 of E-1, a transition to ES-1.1 cannot be made due to RCS pressure conditions. At step #13, a transition to ES-1.2 will be performed.

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Incorrect – See above explanation
D. Correct – See above explanation

Technical Reference(s):
1. ES-1.2, POST LOCA Cooldown and Depressurization, Rev 18
2. E-1, Loss of Reactor or Secondary Coolant, Rev 19

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #8, Objective I: STATE and EXPLAIN the parameters which are evaluated, including their Criteria and Basis, to transition from E-1 to other procedures.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam _____N/A_______

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:
SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	1		
Steam Generator Tube Rupture	**Group**	1		
	K/A	038 G2.4.21		
	Importance Rating	4.6		
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				

Question # 77

The plant experienced a Steam Generator Tube Rupture 3 hours ago.

Current plant conditions are as follows:

- The RCS is at 360°F and is being cooled down at 40°F/hr.
- PZR level is 25% and slowly rising.
- Ruptured S/G cooldown is in progress using the backfill method.
- Ruptured S/G NR level is 80%.
- All Non Ruptured S/Gs NR levels are 30%.
- Ruptured S/G pressure is 1150 psig.
- All Non Ruptured S/G pressure are 1020 psig.
- SR count rate is 200 cps and slowly rising.

Based on the above conditions, a YELLOW path exists in….

A. Integrity

B. Inventory

C. Heat Sink

D. Subcriticality

Answer: D

Explanation:
Based on the above conditions a Yellow path exists in Subcriticality due to SR count rise (due to the dilution from the backfill method in ES-3.1). Heat Sink is plausible as several SG parameters are mentioned but none exceed the Yellow path threshold. Integrity is plausible as the plant is cooling down quickly due to the SGTR but incorrect as Temperature is equal to 350 (which would be a NO answer in the Integrity flowchart) but still greater than Figure 4b of CSF-1. Inventory is plausible as PZR level is less than 92% but greater than 17%.

A. Incorrect – See above explanation
B. Incorrect – See above explanation
C. Incorrect – See above explanation
D. Correct – See above explanation

Technical Reference(s):
1. ES-3.1, Post-SGTR Cooldown using Backfill, Rev 12
2. CSF-1 CSF Status Trees, Rev 13

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #1, Objective K&H:
H. List the six critical safety functions and state which barriers each safety function is associated with.

K. Explain how challenges to critical safety functions are prioritized within each critical safety function.

T61.003D, Emergency Operations, LP #29, Objective B: DESCRIBE the Symptoms and/or Entry conditions for:
1. FR-S.1, Response to Nuclear Power Generation/ATWS.
2. FR-S.2, Response to Loss of Core Shutdown.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:
SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures **YES**
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
055 – Station Blackout	Group #	1		
	K/A #	055 EA2.03		
	Importance Rating	3.9		
Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power.				

Question #78

The plant is in MODE 6 with fuel movement in progress:

- ‘B’ ESF Emergency Bus, NB02, is de-energized for bus cleaning
- A loss of offsite power occurs
- ‘A’ Emergency Diesel Generator (EDG), NE01, failed to automatically start and cannot be started from the Control Room
- The crew has just entered OTO-EJ-00001, Loss of RHR Flow.

What procedure section or FSG should the Control Room Supervisor direct FIRST?

A. FSG-4, ELAP DC Load Shed / Management

B. FSG-5, Initial Assessment and FLEX Equipment Staging

C. OTO-EJ-00001, Attachment D, Restoration of NB01 and NB02

D. OTO-EJ-00001, Attachment C, Loss of RHR Not in Reduced Inventory with Loss of NB01 and NB02

Answer: D

Explanation:

Per OTO-EJ-00001, step 4 RNO 4.a "IF in Mode 5-6, AND RCS level is GREATER THAN 64 inches, THEN Go To Attachment C, Loss Of RHR Not In Reduced Inventory With Loss of NB01 And NB02." is correct as that is the first attachment that should be directed for these plant conditions (no AC power and RPV level greater than 64 inches).

The plant is in a station blackout in MODE 6 and Attachment C will direct actions for power restoration after containment evacuation is directed making it a K/A match.

Attachment D is not the first directed but is plausible as step #4 b RNO 4.b directs Attachment D.

The FSG distractors are from OTO-NB-00005 which is entered from OTO-EJ-00001 step 4 RNO

C.4. FSG -5 is direct from OTO-NB-00005 continuous action step 7 RNO. This is supported by the stem report of offsite power not available for 4 hours making an ELAP declaration plausible and then FSG-5 would be the next action. FSG-4 is directed for OTO-NB-00005 step #10 and is plausible for the same reason as FSG-5

A. Incorrect. See above explanation
B. Incorrect. See above explanation
C. Incorrect. See above explanation
D. Correct. See above explanation

Technical Reference(s):
1. OTO-EJ-00001, Loss of RHR Flow, Rev 038
2. OTO-NB-00005, Loss of ALL AC Power While on RHR, Rev 3

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP D-01, Obj Y, Explain how initial plant conditions affect the applicability of the EOP's.

Question Source: Bank # ______
Modified Bank # _____
New ___X____

Question History: Last NRC Exam __N/A____

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:
k/a match as the question examines the correct procedure to direct to restore power based on plant conditions.

Question written for other than modes 1-4 to prevent overlap with operating exam.

SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures-

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	1		
Loss of Offsite Power	**Group #**	1		
	K/A #	00056 G2.2.25		
	Importance Rating	4.2		
Knowledge of the bases in Technical Specifications for limiting conditions for operation and safety limits				

Question # 79

Per 3.8.1 Technical Specifications Basis, when BOTH Offsite Circuits are inoperable, why is a completion time of 24 hours to restore one offsite circuit to operable status allowed?

A. all redundancy in the AC electrical power supplies has been lost.

B. there is insufficient AC sources available to power the minimum required ESF functions.

C to allow time to evaluate and repair any inoperability of required redundant safety features.

D. the configuration of the redundant AC electrical power systems that remain available is NOT susceptible to a single failure.

Answer: D

Explanation:

A. Incorrect – This is the basis of 3.8.1 Condition H for the condition when 3 or more AC sources are inoperable. Page 3.8.1 -14
B. Incorrect – This is the basis of 3.8.1 Condition E for the condition when 2 EDGS are inoperable. Page 3.8.1 -13
C. Incorrect – This is the basis of the completion time and required action in 3.8.1 Condition A.2 and C.1 for the "required redundant features that are inoperable" section. Its basis states "to allow the operator time to evaluate and repair any discovered inoperabilities" The wording was paraphrased to make it creditable. While this action appears in C.1 (BOTH offsite circuits inoperable), this is action C.1 with a 12 hour from discovery completion time which is not what the question stem is asking (i.e. is asking about the 24 hr clock) therefore making it wrong.
D. Correct – This is the basis of 3.8.1 Condition C for the condition when 2 Offsite circuits are inoperable (i.e Loss of offsite power). Page B 3.8.1-11.

Technical Reference(s):

1. Technical Specification 3.8.1 and it bases

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #1 Objective G: DESCRIBE the two offsite independent circuits which satisfy the Tech Specs offsite power requirements.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(2)

Comments:

k/a match as the question asks the basis for BOTH offsite circuit operable but written in the positive voice. The LCO 3.8.1 C for BOTH offsite circuit inoperable is in effect the "loss of offsite power" and the question examines the TS basis of LCO 3.8.1 Condition C.

SRO ONLY due to the knowledge of Facility operating limitations in the technical specifications and their bases.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #1	1		
Loss of Instrument Air	**Group #1**	2		
	K/A #065 AA2.08	065 AA2.08		
	Importance Rating	3.3		
Ability to determine and interpret the following as they apply to the loss of instrument air: Failure modes of air operated equipment				

Question # 80

The plant is in MODE 2 at NOP/NOT when the RO reports Instrument Air pressure is 60 psig and lowering. The crew has just entered OTO-KA-00001, Partial or Total Loss of Instrument Air.

(1) RCS temperature will be controlled manually using ____(1)______.

And

(2) The CRS should direct which attachment of OTO-KA-00001 NEXT?

A. (1) Condenser Steam Dump Valves
(2) Attachment H, Air Operated Valves Outside Containment

B. (1) Condenser Steam Dump Valves
(2) Attachment D, Reduce Charging Flow to RCP Seal Without Instrument Air

C. (1) Steam Generator Atmospheric Steam Dump Valves
(2) Attachment H, Air Operated Valves Outside Containment

D. (1) Steam Generator Atmospheric Steam Dump Valves
(2) Attachment D, Reduce Charging Flow to RCP Seal Without Instrument Air

Answer: D

Explanation:

Steam dump valves fail shut on loss of IA therefore temperature will have to be controlled used SG ASDs since they have a N2 backup.. At step #6 RNO of OTO-KA-00001, Attachment D is directed and is correct. If Instrument air pressure is 60 psig, instrument air is not available in the Aux building and the RNO will apply. Attachment H is plausible as it is directed in step #10 and step #12, but is not the first attachment directed.

A. Incorrect – See above explanation.
B. Incorrect – See above explanation.
C. Incorrect – See above explanation.

D. Correct

Technical Reference(s):
1. OTO-KA-00001, Partial or Total Loss of Instrument Air, Rev 28

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, Objective C: Given a set of plant conditions or parameters indicating a Partial or Total Loss of Instrument Air, IDENTIFY the correct procedure(s) to be utilized and OUTLINE the high level actions to stabilize the plant

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:
K/A match as the first part of the question tests the " Ability to determine and interpret Failure modes of air operated equipment" while the second part of the question is the SRO component of the question.

SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following?
- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **YES**

- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	Group #	1		
	K/A #	W/E 05 G 2.4.6		
	Importance Rating	4.7		
Knowledge of EOP mitigation strategies.				

Question # 81

A Reactor Trip and Safety Injection have occurred.

- The crew transitioned to FR-H.1, Response to Loss of Secondary Heat Sink
- ‘A’ condensate pump is running with discharge pressure of 320 psig

Per FR-H.1, what EOP Addendum(s) should the CRS direct to restore MFW?

A. ONLY EOP Addendum 30, Establishing Main Feedwater Flow

B. ONLY EOP Addendum 28, Placing the Condensate System in Service AND EOP Addendum 30, Establishing Main Feedwater Flow

C. ONLY EOP Addendum 29, FWIS Bypass Operation AND EOP Addendum 30, Establishing Main Feedwater Flow

D. EOP Addendum 28, Placing the Condensate System in Service, EOP Addendum 29, FWIS Bypass Operation, AND EOP Addendum 30, Establishing Main Feedwater Flow

Answer: C

Explanation:
Per FR-H.1 Step 7 “Try to Establish Main Feedwater flow to at Least one SG”
The first sub set is to check condensate system in service. Based on the conditions in the stem of the question this is met so the CRS should not direct the performance of EOP Add 28. The candidate need to understand what it means “Check condensate system in service. This is met with a condensate pump running and discharge pressure greater 300 psig. Per the procedure a bypass of the FWIS is required per EOP Addendum 29 (This is a plant specific exception to the EOP due to the design of the FWIS). After the FWIS is bypassed Main Feedwater is Established using EOP Addendum 30, Establishing Main Feedwater Flow.

A. Incorrect, Per above, Plausible if the candidate does not understand the plant specific design the FWIS must be bypassed prior to restoring MFW

B. Incorrect, per above. Plausible if the candidate incorrectly believes that since the condensate pump discharge pressure is below the 120B annunciator setpoint of 325 psig that condensate is not considered “in service” and believes that EOP addendum 28 must be completed prior to restarting the MFW system. Only 300 psig is required for the Condensate system to be considered “in service”. Also the FWIS needs to be bypassed prior to restoring MFW.
C. Correct, per above
D. Incorrect, per above. Plausible if the candidate incorrectly believes that since the condensate pump discharge pressure is below the annunciator setpoint of 325 psig that condensate is not considered “in service” and believes that EOP addendum 28 must be completed prior to restarting the MFW system.

Technical Reference(s):
1. FR-H.1, Response to Loss of Secondary Heat Sink, Rev 18
2. OTA-RK-00027 Add 120B, MFP A Suction Pressure Low, Rev 1

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #-26 FR-H.1/FR-H.2/FR-H.3/FR-H.4/FR-H.5, FRG HEAT SINK (H) SERIES Objective I. OUTLINE procedural flowpath including major system and equipment operation in accomplishing the goal of:
1. FR-H.1, Response To Loss Of Secondary Heat Sink.

Question Source: Bank # __X – no bank id yet____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2016 ILT NRC Exam Q#81________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following? **YES**

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **YES, The SRO candidate must select the correct EOP addendum.**
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	1		
Loss of Source Range Nuclear Instrumentation	Group	2		
	K/A	032 G2.1.20		
	Importance Rating	4.6		
Ability to interpret and execute procedure steps.				

Question # 82

Unit is in MODE 5 performing OTG-ZZ-0007, Refueling Preparation, Performance And Recovery.

- Gamma-metrics Monitor N60 is OPERABLE
- Gamma-metrics Monitor N61 is OPERABLE
- Maintenance has just been completed on Source Range N31 and the detector is Bypassed waiting on post maintenance testing to be completed
- Source Range N32 is OPERABLE

Then, Source Range N32 fails LOW.

The CRS entered OTO-SE-00001, Nuclear Instrument Malfunction, and proceeds to Attachment C, Source Range Nuclear Instrument Malfunction.

The Control Room Supervisor will direct what action NEXT?

A. Perform OSP-SF-00001, Shutdown Margin Calculation

B. Perform OSP-BL-00001, Unborated Water Source Isolation Valves/MODE 6

C. Perform OSP-SF-00500, Control Rods Incapable Of Withdrawal

D. Perform OSP-SH-00001, Gamma-metrics Nuclear Instrument Channel Check

Answer: A

Explanation:

A CORRECT: Step C10 RNO directs performing SDM calculation first
B: INCORRECT: SDM Calculations is performed first followed by OSP-BL-00001 in step C10 RNO
C: INCORRECT: Step C13 RNO but not done since two SR intruments inoperable
D: INCORRECT: Extended range Nuclear Instruments can be used to meet the requirements of TS 3.9.3 but still need 2 channels operable to meet the Tech Spec for this mode

Technical Reference(s):
1. OTO-SE-00001, Nuclear Instrumentation Malfunction, Rev 32

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, Objective D: Given a set of plant conditions or parameters indicating a Nuclear Instrument Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ____
New __x____

Question History: Last NRC Exam ____N/A__

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:
SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following?
- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps **YES**
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
Loss of Condenser Vacuum / 4	**Group #**	2		
	K/A #	051 AA2.02		
	Importance Rating	4.1		
Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip.				

Question # 83

Reactor Power is 8%.

- Main Turbine startup is in progress at 800 rpm.
- Annunciator 116B, Cond A Vac Lo is LIT.
- The crew has entered OTO-AD-00001, Loss of Condenser Vacuum.
- Boron Equalization has been placed in service per step 9.
- The BOP reports that Condenser Backpressure is 9 inches HGA and stable.

Based on the above conditions, what is the correct procedural flowpath to implement?

A. E-0, Reactor Trip or Safety Injection, and then transition to ES-0.1, Reactor Trip Response.

B. OTO-AC-00001, Turbine Trip below P-9, and then transition to OTG-ZZ-00003, Plant Startup Hot Zero Power to 30% Power – IPTE.

C. OTO-AC-00001, Turbine Trip below P-9, and then perform Attachment D, Manually Transferring From MFRVs To MFRV Bypass Valves.

D. OTO-AC-00001, Turbine Trip below P-9, and then transition to OTG-ZZ-00005, Plant Shutdown 20% to Power to Hot Standby.

Answer: D

Explanation:

Note: The setpoint of the condenser Lo Vacuum alarm is 6.5 psig and the automatic turbine trip setpoint is at 8.5 in hgA.

Based on condenser backpressure and power level, only a turbine trip is required per step #1 of OTO-AD-00001. Per continuous action step #1 and its RNO, If reactor power was greater than 10 % a manual reactor trip would have been required which makes E-0 and ES-0.1 plausible. If it

is understood that a turbine trip is required but continued power operation is allowed at 8%, OTO-AC-00001 to OTG-ZZ-00003, Plant Startup Hot Zero Power to 30% Power to IPTE is plausible, Per OTG-ZZ-00003, Step 2.4 - "This procedure contains instructions for raising Reactor power from 10-8 Amps to 30% RTP and has been evaluated for the following consequences:

- *Abnormal conditions during Main Turbine roll to 1800 rpm to avoid potential damage to the turbine.*
- *Main Turbine excessive time at 1800 rpm to avoid overheating due to the lack of final stage cooling.*
- *Attaining too low a Steam Generator (SG) level prior to transferring from the Bypass MFRVs to the MFRVs to avoid a low Steam Generator level Reactor Trip."*

Per OTO-AC-00001 step #21, the following are possible choices for transition from OTO-AC-00001:

"21. Refer To Applicable Portions Of The Following Procedures To Align The Plant For Current Plant Conditions:"

- *IF Required Emergency Purge Hydrogen gas from Main Generator per OTN-CC-00001, Main Generator Gas System And Gas System Auxiliaries*
- *OTG-ZZ-00004, Power Operation*
- *OTG-ZZ-00005, Plant Shutdown 20% Power To Hot Standby*
- *OTN-AC-00001, Main Turbine And Generator Systems*
- *OTN-AD-00001, Condensate System*
- *OTN-BG-00002, Reactor Makeup Control and Boron Thermal Regeneration System*

OTN-AC-00001, Add 3 is incorrect as its purpose is to prepare for a planned turbine generator trip and NOT response to a turbine generator trip

A. Incorrect – see above explanation – the reactor does not trip and therefore the transition to E-0 and ES-0.1 are not correct.
B. Incorrect – see above explanation – the power levels are incorrect for OTG-ZZ-00004
C. Incorrect - Attachment D is olyn performed if Automatic transfer did not occur
D. Correct - see above explanation

Technical Reference(s):
1. OTO-AD-00001, Loss of Condenser Vacuum, Rev 34
2. OTA-RK-00026 ADD 116B, Condenser A Vacuum Lo, Rev 0
3. E-0, Reactor Trip or Safety Injection, Rev 21
4. OTO-AC-00001, Turbine Trip below P-9, Rev 25
5. OTG-ZZ-00003, Plant Startup Hot Zero Power to 30% Power – IPTE, Rev 63
6. OTG-ZZ-00005, Addendum 3, Maintaining Mode 1 with the Turbine Tripped, Rev 6
7. OTN-AC-00001, Main Turbine and Generator Systems, Rev 51
8. OTN-AC-00001, Addendum 3, Turbine Generator Shutdown, Rev 15

References to be provided to applicants during examination: None

Learning Objective: T61.003B, Off Normal Operations, LP#7, Objective D & G:

D. Given a set of plant conditions or parameters indicating a Loss of Condenser vacuum, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

G. RECOGNIZE the conditions that would require a Reactor Trip / Turbine Trip.

Question Source: Bank # ___X – no bank id yet___
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? NO

Can the question be answered solely by knowing immediate operator actions? NO

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? NO

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? NO

Does the question require one or more of the following? YES

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures **YES**
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures **YES**

k/a match as the question asks what is the correct procedural flowpath to implement (all of which have a reactor and/or reactor trip) to place the plant in a safe condition given the event in progress. Knowledge of the procedural network plus it rules for implementation are required making the question SRO knowledge level and therefore the k/a apply to SROs. The addition of the sub procedure (ES-0.1 or the OTG) in the possible choices mets the intent of more than just the entry level condition into AOPs or major E-0 (RO knowledge level) making the question SRO per the above comment of Figure 2.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	1		
Loss of Containment Integrity	**Group**	2		
	K/A	069 G2.4.41		
	Importance Rating	4.6		
Knowledge of the emergency action level thresholds and classifications				

(REFERENCE PROVIDED)

Question # 84

A manual Reactor Trip and Safety Injection were performed based on the following conditions:

- Loss of Offsite Power
- Unexpected rise in "A" SG narrow range level
- Lowering PZR level and pressure
- Condenser radiation monitor in HIGH alarm
- "A" Main Steamline radiation monitors in HIGH alarm

Steam Generator "A" pressure is noted to be lower than expected and an automatic MSLIS actuation is received. A report from the field indicates one of the Steam Generator "A" safety valves is blowing steam.

Chemistry reports that Dose Equivalent (DE) I-131 is 280 µci/cc.

The Emergency Action Level classification that will be reported to Sentry is...

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C

Explanation:

A INCORRECT: UE for loss of offside power
B: INCORRECT: Alert for SGTR only
C: CORRECT: SAE for Loss of RCS and Loss of Containment

D: INCORRECT: DE I-131 needs to be > 300 µci/cc for GE

Technical Reference(s):

1. EIP-ZZ-00101, Classifications of Emergencies, Rev 55
2. EIP-ZZ-00101, Add 1 Emergency Action Level Classification Matrix

References to be provided to applicants during examination: EIP-ZZ-00001, Add 1 (wallchart size)

Learning Objective: Lesson T68.1020.6 (.8), Obj B, Determine the emergency classification for given indications and/or symptoms, per EIP-ZZ-00101.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:
SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
RCS Overcooling—Pressurized Thermal Shock	Group #	2		
	K/A #	W/E08 EA2.1		
	Importance Rating	4.2		
Ability to determine and interpret the following as they apply to the (Pressurized thermal Shock): Facility conditions and selection of appropriate procedures during abnormal and emergency conditions				

Question # 85

The crew has transitioned to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition.

The following plant conditions exist:

- RCS pressure is 1000 psig and stable
- Pressurizer Level is 90% and rising
- All RCS cold leg temperatures are 190°F and lowering slowly
- Containment Pressure is 10 psig and stable
- SG Pressures are as follows:
    - 'A' & 'B' S/Gs are 200 psig and lowering
    - 'C' S/G is 600 psig and lowering slowly
    - 'D' S/G is 800 psig and stable
- RVLIS pumps off indication is 100%
- CST to AFP suction header pressure is 11 psig and slowly lowering

What Procedure or EOP Addendum should the Control Room Supervisor direct NEXT?

A. EOP Addendum 3, Starting An RCP

B. EOP Addendum 5, Establishing Excess Letdown

C. EOP Addendum 19, Aligning ESW to AFW Suction

D. ECA-2.1, Uncontrolled Depressurization of All Steam Generators

Answer: B

Explanation:

Based on the saturation pressure of 1000 psig, saturation temperature is ~545F. With the values given in the stem, RCS subcooling is greater than 200F

A. Incorrect – Per FR-P.1 steps #6 and #13, the RNO which direct EOP ADD 3 does not apply and conditions are not met to start an RCP. Step #13 is a continuous action step and applicable anytime after it has been reached.
B. Correct – per step #21 RNO actions, the PZR level needs to be lowered and directing EOP ADD 5 is correct. This action is important as this will aid in R CS pressure reduction to exit the FR-P.1 Red path.
C. Incorrect but plausible as the header suction pressure has lowered to less less than the EOP addendum 42 criteria but is still higher than the 3.5 EOP Addendum 19 setpoint. These actions are on the foldout page of FR-P.1 and are applicable at any time.
D. Incorrect but plausible based on the fact that a way to arrive at FR-P.1 is to have multiple S/G depressurizing at the same time. This would provide the RCS cooldown but allow the RCS to stay pressurized to some extent considering SI flow. Incorrect as there is no transition from FR-P.1 to ECA-2.1. ECA 2.1 is entered from E-2 but the candidate may believe this procedure is necessary to prevent the continued cooldown as A& B S/Gs show indications of a fault.
FR-P.1 Step #2 states to minimize cooldown from faulted S/Gs which adds to the plausibility of this distractor but no yet again there is no transition in FR-P.1.

Technical Reference(s):
1. FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, Rev 13
2. CSF-1, Critical Safety Function Status Trees, Rev 13

References to be provided to applicants during examination: None

Learning Objective: T61.003D, Emergency Operations, LP #28 Objective G “STATE and EXPLAIN the parameters which are evaluated, including their Criteria and Basis, to transition from the following procedures to another procedure.
1. FR-P.1, Response To Imminent Pressurized Thermal Shock Condition.”

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(5)

Comments:

SRO ONLY due to ES401 Figure 2 of NUREG 1021 as follows:

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed **YES**
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
005 Residual Heat Removal	Group #	1		
	K/A #	005 G2.2.38		
	Importance Rating	4.5		
Knowledge of conditions and limitations in the facility license				

Question # 86

Per Technical Specifications Refueling Basis, the purpose of at least one RHR loop being Operable and in operation is to …..

A. Support filling or draining of the refuel pool.

B. Provide mixing of borated coolant to minimize the possibility of criticality.

C. Allows operations such as core mapping or RCS to RHR isolation valve testing.

D. To ensure the iodine release due to a postulated fuel handling accident are within the limits of 10 CFR 100.

Answer: B

Explanation: Per the Technical Specification Bases of 3.9.5 - At least one RHR loop must be Operable and in operation to provide:

a. Removal of Decay heat
b. Mixing of borated coolant to minimize the possibility of criticality
c. Indication or reactor coolant temperature.

A. Incorrect. Plausible as TS basis 3.9.6 describes the purpose of the standby RHR train not the train in operation.
B. Correct. Per TS basis 3.9.5 and 3.9.6, this is one of the 3 purposes of RHR in the Refueling TS.
C. Incorrect. Plausible as this is the reason for the NOTE in the LCO that allows RHR to be secured for 1 hour every 8 hours
D. Incorrect. This is purpose of the 23 feet above the top of the RPV flange which is in the applicability section of TS 3.9.5. The 23 feet requirement is further described in the T.S 3.9.7 and its basis.

Technical Reference(s):
1. Technical Specification 3.9.5 and its bases

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Learning Objective:

1. T61.0110 Systems, LP#7 Residual Heat Removal, Objective B

   "DESCRIBE the purpose and operation of the following RHR System components, to include interlocks, controller operation and power supplies."

   1. RHR pumps
   2. RHR Heat Exchangers (HXs)
   3. Reactor Coolant System (RCS) Hot Leg Suction Valves to RHR
   4. Containment (CTMT) recirculation (Recirc) Sump Suction Valves to RHR
   5. Refueling Water Storage Tank (RWST) Suction Valves to RHR
   6. RHR Heat Exchanger Flow Control Valves
   7. RHR Heat Exchanger Bypass Valves
   8. RHR Bypass Miniflow Valves
   9. RHR Discharge Valves to the Safety Injection (SI) and Centrifugal Charging Pumps
   10. Suction Relief Valves
   11. RHR Cold Leg Injection Valves
   12. RHR SI System Hot Leg Recirc Isolation Valves
   13. RHR Hot Leg Injection Isolation Valve
   14. RHR Downstream Relief Valves

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A__

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

10 CFR Part 55 Content:

10 CFR:55.43(b)(2)

Comments:

SRO only due to Facility operating limitations in the Technical Specifications and their bases.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	2		
Pressurizer Pressure Control	Group	1		
	K/A	010 G2.2.25		
	Importance Rating	4.2		
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits				

Question # 87

Reactor Power is 90% at EOL.

- RCS Boron concentration is 500 ppm.
- PZR Boron concentration is 600 ppm.
- Pressurizer Pressure Channel, BB PT-457, is controlling Pressurizer pressure.
- BB PT-457 fails **LOW**.

(1) With NO operator action, RCS average temperature will _____(1)_____?

And

(2) Why do Technical Specifications allow 1 hour to verify P-11 is in the correct state?

A. (1) lower
(2) to ensure shutdown actions are initiated in the event of a complete loss of ESFAS function

B. (1) lower
(2) to ensure protection against violating the DNBR limit

C. (1) raise
(2) to ensure shutdown actions are initiated in the event of a complete loss of ESFAS function

D. (1) raise
(2) to ensure protection against violating the DNBR limit

Answer: A

Explanation:

With Pressurizer Boron concentration higher than RCS boron concentration, the given failure results in a higher PZR Pressure and therefore an outsurge of higher borated water into the RCS. Thus the higher RCS boron concentration will results in a lower RCS temperature.

Per Tech Spec bases page B3.3.2-59 condition L applies to the P-11 Interlock and "the 1 hour completion time is equal to the time allowed by LCO 3.0.3 to 'initiate shutdown actions in the event of a complete loss of ESFAS function'. The reason of "provide protection against violating the DNBR limit" is technical specification bases for the RTS Instrument Pressurizer Pressure Low.

A. Correct
B. Incorrect - wrong reason
C. Incorrect - wrong RCS temperature direction
D. Incorrect - both are wrong

Technical Reference(s):.
1. OTO-BB-00006, Pressurizer Pressure Control Malfunction,
2. Technical Specification and their bases Section 3.3.1 and 3.3.2

References to be provided to applicants during examination: None

Learning Objective: T61.003B6, Off normal operations, LP #41, Objective C. Given a set of plant conditions or parameters indicating a Pressurizer Pressure Control Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ______N/A______

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

10 CFR Part 55 Content:

10 CFR:55.43(b)(2)

Comments:

SRO Only due to Facility operating limitations in the Technical Specifications and their bases

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	2		
Reactor Protection	**Group**	1		
	K/A	012 A2.03		
	Importance Rating	3.7		
Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Incorrect channel bypassing				

Question # 88

Reactor Power is 100%.

- Pressurizer Pressure Channel, BB PT-457, fails low.
- The crew enters OTO-BB-00006, Pressurizer Pressure Control Malfunction, to respond to the failure and stabilize the plant.
- When tripping bistables for Channel 457 IAW OTO-BB-00006, bistables for Channel BB PT-456 are inadvertently tripped due to improper communication.

(1) What is the reportability for this event?

And

(2) What Emergency Operation Procedure will be implemented after E-0, Reactor Trip or Safety Injection, is complete?

A. (1) Reportable
(2) ES-0.1, Reactor Trip Response

B. (1) NOT reportable
(2) ES-0.1, Reactor Trip Response

C. (1) Reportable
(2) ES-1.1, SI Termination

D. (1) NOT reportable
(2) ES-1.1, SI Termination

Answer: C

Explanation:

A. Incorrect. Plausible as this response for procedure implementation would be correct if operator does not recognize that an SI has occurred. Also, reportability requirement is correct.
B. Incorrect. Plausible as this response for procedure implementation would be correct if operator does not recognize that an SI has occurred. Reportability portion is not correct.
C. Correct. An SI will occur since two PZR pressure channels will have low bistables tripped. Any ESF signal is reportable.
D. Incorrect. Plausible response as procedure implementation is correct. Incorrect due to second part concerning reportability. Even if operator induced an event is reportable.

Technical Reference(s):
1. E-0,Reactor Trip or Safety Injection, Rev 21
2. APA-ZZ-00520, Reporting Requirements And Responsibilities, Rev 50
3. 7250D64, Sheet 6, Pressurizer Trip Signals, Rev 6

References to be provided to applicants during examination: None

Learning Objective: T61.003D, LP D-4, Objective L, Outline procedural flowpath including major system and equipment operation in accomplishing the goal of E-0.

T61.0110, LP 69, Objective D.2, Perform the following as they pertain to APA-ZZ-00520, “Reporting Requirements and Responsibilities”: 2. Discuss the incidents reportable in the following time frames: e. Immediate (4 hours).

Question Source: Bank # ___X___
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2013 Q#86_

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(1)

Comments:
SRO level question based on SRO knowledge required for reportability criteria which is a Conditions and limitation of the facility license

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
Engineered Safety Feature Actuation	Group #	1		
	K/A #	013 G2.2.36		
	Importance Rating	4.2		
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.				

Question # 89

Reactor Power is 100% when the power supply for ESFAS Channel 2 becomes degraded.

Channel 2 of the ESFAS Cabinet is being de-energized for maintenance in accordance with Attachment 4, De-Energizing Channel II ESFAS Cabinets, of OTS-SA-00001, Operation of Engineered Safety Feature Actuation System, when Annunciator 128B, TD AFP START, alarms.

(1) Is this an expected alarm?

And

(2) Which of the following functions is made INOPERABLE by de-energizing Channel 2 ESFAS?

A. (1) Yes
(2) Turbine Trip on SG high water level.

B. (1) Yes
(2) Auxiliary Feedwater Pump Suction Transfer on low suction pressure.

C. (1) No
(2) Turbine Trip on SG high water level.

D. (1) No
(2) Auxiliary Feedwater Pump Suction Transfer on low suction pressure.

Answer: B

Explanation:
OTS-SA-00001, Precautions and Limitations
3.2. De-energizing Channel II

3.2.1. The Technical Specifications that are applicable when de-energizing Channel II ESFAS or when a Channel II power supply fails are provided in Attachment 6.
3.2.2. T/S LCO 3.3.2 should be reviewed for Channel OPERABILITY and MODE restraints.
3.2.3. T/S LCO 3.6.3 should be reviewed for Containment penetrations.
3.2.4. An AFW Low Suction Pressure Signal will be initiated.

Per Attachment 4, De-Energizing Channel II ESFAS Cabinets, of OTS-SA-00001, Operation of Engineered Safety Feature Actuation System Annunciator 128B, TD AFP START, will alarm.
Per OTS-SA-00001, Attachment 4 references TS 3.3.2-6 A & H, when de-energizing ESFAS Channel 2
Per T.S. 3.3.2-6.A and H, AFAS Manual Initiation and Auxiliary Feedwater Pump Suction Transfer on low suction pressure functions are INOPERABLE.

A. Incorrect – This alarm will actuate, however the Turbine Trip on SG high water level function is INOPERABLE if Channel 1 or 4 is de-energized
B. Correct - This alarm will actuate, Per OTS-SA-00001, Operation of Engineered Safety Feature Actuation System, De-energizing makes Auxiliary Feedwater Pump Suction Transfer on low suction pressure functions INOPERABLE per T.S.3.3.2-6.H
C. Incorrect - This alarm will actuate however the Turbine Trip on SG high water level function is INOPERABLE if Channel 1 or 4 is de-energized
D. Incorrect - This alarm will actuate Per OTS-SA-00001, Operation of Engineered Safety Feature Actuation System, De-energizing only make the AFAS Manual Initiation and Auxiliary Feedwater Pump Suction Transfer on low suction pressure functions INOPERABLE.

Technical Reference(s):
1. OTS-SA`-00001, Operation Of Engineered Safety Feature Actuation System, Rev 19
2. TS and Bases 3.3.2, ESFAS Instrumentation

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #52 EFSAS, Objective F. DISCUSS the purpose and scope of the following:
1. OTS-SA-00001, "De-energizing and Energizing Engineered Safety Feature Actuation System".

Question Source: Bank # __X____
Modified Bank # ______
New _______

Question History: Last NRC Exam __2014 Q#88__

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

10 CFR Part 55 Content:

10CFR 55.43(b)(2)

Comments:

SRO Only due to Facility operating limitation and Technical Specifications and their bases.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	2		
Containment Cooling	**Group**	1		
	K/A	022 G2.2.22		
	Importance Rating	4.7		
Knowledge of limiting conditions for operations and safety limits.				

(REFERENCE PROVIDED)

Question # 90

Reactor Power is 100%.

- "A", "C", and "D" Containment Cooling Units are in service
- The "D" unit develops high vibration, is declared "inoperable", and is secured after "B" unit is started
- Five minutes after starting the "B" unit, it trips on overcurrent
- A local reset is attempted but the "B" unit will not start

What are the applicable Technical Specifications for these events?

A. Restore containment cooling train to Operable status within 7 days And 10 days from discovery of failure to meet the LCO **AND** analyze samples of the containment atmosphere within 24 hours **AND** restore required containment atmosphere particulate radioactivity monitor to Operable status within 30 days.

B. **ONLY** Restore containment cooling train to Operable status within 7 days **AND** 10 days from discovery of failure to meet the LCO.

C. **ONLY** Analyze samples of the containment atmosphere within 24 hours **AND** restore required containment atmosphere particulate radioactivity monitor to Operable status within 30 days.

D. Be in Mode 3 in 6 hours **AND** Mode 5 in 36 hours.

Answer: A

Explanation:

Per the Precaution and Limitation #3.13 in OTN-GN-00001, Containment Cooling and CRDM Cooling, If SGN01D, CTMT COOLER UNIT D, is turned off in MODES 1 through 4, T/S 3.4.15 Actions should be entered for containment atmosphere particulate radioactivity monitors GTRE0031 and GTRE0032.

Therefore, Technical Specification 3.4.15 condition B will be declared not met and the required actions of B.1.1, Analyze samples of the containment atmosphere ONCE per 24 hours OR B.1.2 are required AND B.2.1. Restore required containment atmosphere particulate radioactivity monitor to OPERABLE status within 30 days OR B.2.2 are required. In addition to these 3.4.15 actions, Tech Spec 3.6.6 Condition C is not met. Required action C.1 is required within 7 days AND 10 days from discovery of failure to meet the LCO.

Per TS 3.6.6 Two containment cooling trains are required to be operable. Per TS Bases 3.6.6 a train of containment cooling includes cooling coils, dampers, two fans, instruments and controls.

Based on the Tech spec action statements for the conditions given, Restore containment cooling train to Operable status within 7 days And 10 days from discovery of failure to meet the LCO AND analyze samples of the containment atmosphere within 24 hours And restore required containment atmosphere particulate radioactivity monitor to Operable status within 30 days.

A. Correct see explanation above.
B. Incorrect. Both 3.4.15 and 3.6.6 actions need to be performed see explanation above
C. Incorrect Both 3.4.15 and 3.6.6 actions need to be performed see explanation above
D. Incorrect. The containment cooling train with the A and C fan is operable Required action 3.6.6 E is not entered for this situation.

Technical Reference(s):.
1. TS and Bases 3.4.15, RCS leakage detection instrumentation,
2. TS and Bases 3.6.6 Containment spray and cooling system,
3. OTN-GN-00001, Containment Cooling and CRDM Cooling, Rev 030

References to be provided to applicants during examination:
1. Technical Specification LCO 3.4.15
2. Technical Specification LCO 3.6.6

Learning Objective: T61.0110, Systems, LP-40, Containment Ventilation, Objective R, EXPLAIN the precautions, limitations and bases for the following processes/conditions associated with OTN-GN-00001, "Containment and CRDM Cooling"

Question Source: Bank # _X___L16440_____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(2)

Comments:
SRO per 10 CFR 55.43(b)(2) Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1).

Can question be answered solely by knowing ≤ 1 hour TS/TRM Action? **No**
Can question be answered solely by knowing the LCO/TRM information listed "above-the-line?" **No**
Can question be answered solely by knowing the TS Safety Limits? **No**
Does the question involve one or more of the following for TS, TRM, or ODCM?
- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1) **Yes** **SRO-only question**

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. The containment sump level and flow monitoring system;

b. One containment atmosphere particulate radioactivity monitor; and

c. The containment cooler condensate monitoring system.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required containment sump level and flow monitoring system inoperable.	A.1	----------- NOTE ----------- Not required until 12 hours after establishment of steady state operation. ------------------------------	
		Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>		
	A.2	Restore required containment sump level and flow monitoring system to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required containment atmosphere particulate radioactivity monitor inoperable.	B.1.1	Analyze samples of the containment atmosphere.	Once per 24 hours
	<u>OR</u>		
	B.1.2	---------- NOTE ---------- Not required until 12 hours after establishment of steady state operation. ------------------------------	
		Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>		
	B.2.1	Restore required containment atmosphere particulate radioactivity monitor to OPERABLE status.	30 days
	<u>OR</u>		
	B.2.2	Verify containment air cooler condensate monitoring system is OPERABLE.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required containment cooler condensate monitoring system inoperable.	C.1	Perform SR 3.4.15.1.	Once per 8 hours
	<u>OR</u>		
	C.2	----------- NOTE ------------ Not required until 12 hours after establishment of steady state operation. --------------------------------	
		Perform SR 3.4.13.1.	Once per 24 hours
D. Required containment atmosphere particulate radioactivity monitor inoperable. <u>AND</u> Required containment cooler condensate monitoring system inoperable.	D.1	Restore required containment atmosphere particulate radioactivity monitor to OPERABLE status.	30 days
	<u>OR</u>		
	D.2	Restore required containment cooler condensate monitoring system to OPERABLE status.	30 days
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.2	Be in MODE 5.	36 hours
F. All required monitoring methods inoperable.	F.1	Enter LCO 3.0.3.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One containment spray train inoperable.	A.1	Restore containment spray train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	84 hours
C. One containment cooling train inoperable.	C.1	Restore containment cooling train to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. Two containment spray trains inoperable. OR Two containment cooling trains inoperable.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.6.1	------------------------------ NOTE ------------------------------ Not required to be met for system vent flow paths opened under administrative control. --- Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.2	Operate each containment cooling train fan unit for $\geq$ 15 minutes.	In accordance with the Surveillance Frequency Control Program

(continued)

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	2		
Nuclear Instrumentation	Group	2		
	K/A	015 A2.04		
	Importance Rating	3.8		

Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects on axial flux density of control rod alignment and sequencing, xenon production and decay, and boron vs. control rod reactivity changes

(REFERENCE PROVIDED)

Question # 91

Given the following plant conditions:

- 12 hours ago, Shutdown Bank A Rod P-4 dropped into the core
- The rod bottom light for rod P-4 is LIT
- Currently:
  - o Reactor Power is 70%
  - o Computer Point REU1153, AVG RAD LOWER TILT Q3, is in alarm and reading 1.03

(1) Considering Xenon effects ONLY, Power Range NI 41 readings will ________ over the next 36 hours?

And

(2) After I&C corrects the cause of the failure, the CRS will direct what procedure?

A. (1) rise
(2) ESP-ZZ-00004, Flux and Thermocouple Mapping

B. (1) rise
(2) OTO-SF-00001, Attachment B, Dropped / Misaligned Rod Recovery

C. (1) lower
(2) ESP-ZZ-00004, Flux and Thermocouple Mapping

D. (1) lower
(2) OTO-SF-00001, Attachment B, Dropped / Misaligned Rod Recovery

Answer: B

Explanation:
With Shutdown Bank Control rod P-4 fully inserted, a local Xenon transient began 12 hours ago. The Production of Iodine (XE precursor), which is dependent of the number of fissions locally, lowered but so did the burnout of Xenon due to the lower neutron flux. This results in local Xenon concentration spiking over the next 6-10 hours, reaching a peak ~10 hours after the dropped rod. After this point, local Xenon concentration will lower as less Xenon is produced (less Iodine) resulting in less of a local poison concentration. Less poison encourages neutron flux and which the rising flux there is more neutron leakage resulting in more neutrons reaching NI 41. The results would be a NI 41 reading ***rising as the xenon concentration lowers****.*

Due to the malfunction, the crew will be implementing OTO-SF-00001. Specifically, the crew will be at step A11 waiting for I&C to find and fix the cause of the malfunction. ***Upon the report that the problem has been found and corrected, the CRS will direct performance of OTO-SF-00001 Attachment B per step A12.***

ESP-ZZ-00004 is plausible as it is performed concurrently with ESP-ZZ-00006, Incore/Excore Calibration. This is plausible as a quadrant power tilt is occurring from the dropped rod and xenon transient. However, as the Xenon transient is in progress and the prerequisites of ESP-ZZ-00004 require xenon is within 5% of equilibrium, this is not the correct application of this procedure. Correcting the initial problem (dropped rod) is the correct choice which will then correct the QPTR concern.

A. Incorrect – wrong procedure selection
B. Correct
C. Incorrect - both are wrong
D. Incorrect - wrong direction

Technical Reference(s):.
1. Curve Book, Figure 8-7, RCS LOOP with Control Rods and Excore Neutron Detector Locations, Rev. 000
2. OTO-SF-00001, Rod Control Malfunctions,
3. OSP-SE-00003, Quadrant Power Tilt Ration Calculation,
4. OSP-SF-00002, Control Rod Partial Movement,
5. OTA-RK-00022, ADD 81B, Rod at Bottom,
6. ESP-ZZ-00004, Flux and Thermocouple Mapping,
7. ESP-ZZ-00006, Incore/Excore Calibration,

References to be provided to applicants during examination:
1. Curve Book, Figure 8-7, RCS LOOP with Control Rods and Excore Neutron Detector Locations,

Learning Objective: T61.GFES, Reactor Operational Physics, LP #44, Objective 22: Explain reactor response to a control rod insertion.

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Callaway Plant
Senior Reactor Operator

T61.003B 6, Off normal Operations, LP #45, OTO-SF-00001, Objective D: Given a set of plant conditions or parameters indicating a Rod Control Malfunction, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ___X___
Modified Bank # ______
New _______

Question History: Last NRC Exam _2014 SRO Retake__

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(.5)

Comments:

SRO Only because this question involves Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Specifically, per Figure 2 of ES-401,

Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **NO**

Can the question be answered solely by knowing immediate operator actions? **NO**

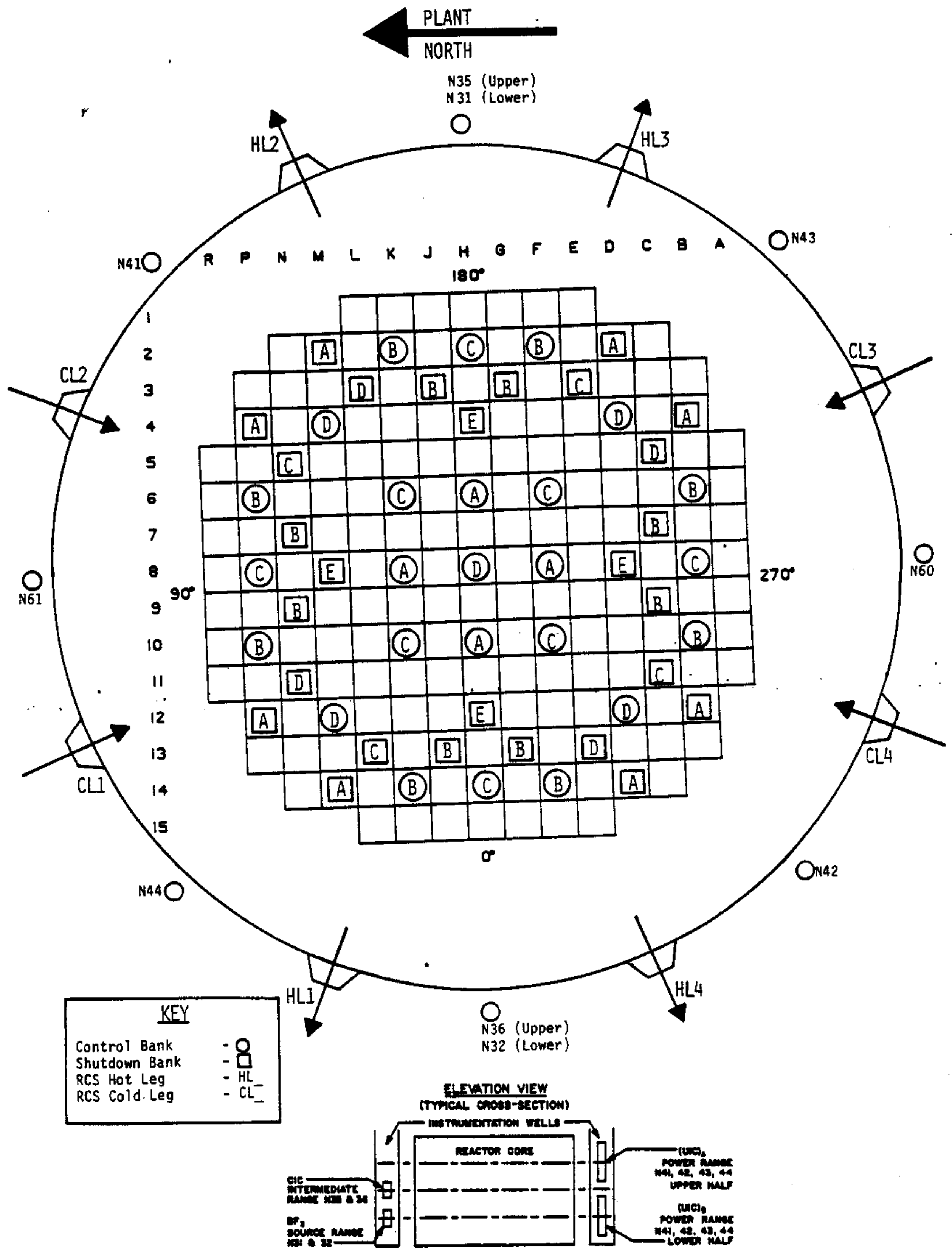
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **NO**

Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **NO**

Does the question require one or more of the following?
• Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
YES SRO-only question

FIGURE 8-7

RCS LOOP ORIENTATION WITH CONTROL ROD AND EXCORE NEUTRON DETECTOR LOCATIONS



12-30-86

Superintendent, Engineering Date

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	2		
Fuel Handling Equipment	Group	2		
	K/A	034 A2.01		
	Importance Rating	4.4		
Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Dropped fuel element				

Question # 92

Refueling is in progress.

- A spent fuel assembly is being moved from the reactor to the upender
- The spent fuel assembly is dropped to the bottom of the canal

What is the sequence of procedural actions and why?

A. Actuate CRVIS, ensure CTMT closure, actuate CPIS; to prevent unfiltered releases out of the equipment hatch.

B. Ensure CTMT closure, actuate CRVIS, actuate CPIS; to minimize dose to personnel inside containment.

C. Actuate CPIS, ensure CTMT closure, actuate CRVIS; to remove airborne Iodine released from dropped assembly.

D. Actuate CRVIS, actuate CPIS, ensure CTMT closure; to ensure control room dose rate is less than 10 CFR 100.

Answer: A

Explanation:
Per OTO-KE-00001, the order of required actions is to actuate CRVIS (step2), ensure CTMT closure (step 12), and actuate CPIS (step 13). CPIS is not manually actuated until the containment is isolated to prevent non-filtered releases out of the equipment hatch which is described in a Note prior to Step #13.

A. Correct – See above explanation
B. Incorrect – Wrong order of actions. The reason is plausible as protecting people in containment is a function of operations personnel
C. Incorrect – Wrong order of actions. The reason is plausible as airborne Iodine is a concern during any fuel handling accident
D. Incorrect – Wrong order of actions. The reason is plausible as accident analysis must very dose to control room operators remains less than 10 CFR 100 limits and is one of the reasons "in vessel" fuel moves must wait 72 hours to commence.

Technical Reference(s):
1. OTO KE-00001, “Fuel Handling Accident”, Rev 16

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems LP #79, Fuel Handling System, Objective I Describe the Purpose, Symptoms or Entry Conditions, and major action steps of OTO-KE-00001, “FUEL HANDLING ACCIDENT.”

Question Source: Bank # _______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A____

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(7) – Fuel Handling facilities and procedures

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 1
	Tier	2		
Control Room Ventilation	Group	2		
	K/A	050 G2.2.40		
	Importance Rating	4.7		
Ability to apply Technical Specifications for a system				

Question # 93

Reactor Power is 100%.

- OTN-GK-00001, Control Building HVAC System section 5.3 was completed to secure SGK04A, 'A' Control Room AC Unit, for maintenance
- SGK04A, compressor and fan, is tagged out and work has commenced
- Control Room temperature is now 84°F and rising 1°F every 5 minutes
- SGK04B is checked and GK-V766, Condenser Cooling Water Circuit Regulating Control Valve, is found fully OPEN

What is the MOST limiting Technical Specification action required?

A. Enter L.C.O 3.0.3 and commence a plant shutdown to MODE 5

B. Enter L.C.O 3.7.10, condition A and restore inoperable CREVS train

C. Enter L.C.O 3.7.10, condition B and restore inoperable CRE/CBE envelope

D. Enter L.C.O 3.7.11, condition A and restore inoperable CRACS train

Answer: A

Explanation:

A. Correct. 2 trains of CRACs OOS CONDITION E of LCO 3.7.11 is entered which requires LCO 3.0.3 entry
B. Incorrect. Plausible because one train of CREVS is OOS due to CRAC maintenance
C. Incorrect. Plausible because the control room temperature is at the limit and it may be thought that this causes the CRE to be inoperable
D. Incorrect. Plausible because one train of CRAC is OOS and it is recognized that other train is not cooling, thus inoperable

Technical Reference(s):
1. Technical Specifications for 3.0.3, 3.7.10 and 3.7.11

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

References to be provided to applicants during examination: None

Learning Objective: T61.010, LP 39, Objective F: STATE the Tech Spec and FSAR LCO's for the Aux Building / Fuel Building HVAC equipment and Area Temperature Monitoring

Question Source: Bank # ______
Modified Bank # ______
New ___X___

Question History: Last NRC Exam __N/A_____

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(2)

Comments:

SRO per 10 CFR 55.43(b)(2) Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1).

Can question be answered solely by knowing ≤ 1 hour TS/TRM Action? **No**
Can question be answered solely by knowing the LCO/TRM information listed "above-the-line?" **No**
Can question be answered solely by knowing the TS Safety Limits? **No**
Does the question involve one or more of the following for TS, TRM, or ODCM?
- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1) **Yes** **SRO-only question**

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Conduct of Operations	Group #	Generic		
	K/A #	G2.1.5		
	Importance Rating	3.9		
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.				

Question # 94

The Plant is in a refueling outage.

- A flaw has been found on the refueling equipment that is adverse to safety
- Repair of this flaw will require 2 maintenance individuals to work continuously until it is repaired
- Maintenance, Operations and Radiation Protection are on a 5 day on 2 day off 12 hour working schedule
- Both of the required maintenance individuals were off the 2 previous days
- The maintenance supervisor has requested a waiver of work hour limitations

(1) Per APA-ZZ-00905, Limitations of Callaway Plant Staff Working Hours, a work hour limitation violation will occur if these individual work greater than __________ without an approved waiver?

And

(2) Per APA-ZZ-00905 and CA0161, Waiver of Work Hour Limits, who should approve the waiver?

A. (1) 12 hours
(2) Shift Manager

B. (1) 16 hours
(2) Shift Manager

C. (1) 12 hours
(2) Director, Nuclear Operations

D. (1) 16 hours
(2) Director, Nuclear Operations

Answer: B

Explanation:
Per Section 3 of form CA0161 and Section 3.4, 3.5 and 4.8 or APA-ZZ-00905, the Shift Manager (or Security Supervisor) will Certify that a waiver is necessary to mitigate or prevent a condition adverse to safety Or Maintain site Security prior to approving the waiver. Per APA-ZZ-00905 section 4.2, step 4.2.1 – the limit on hours worked is 16 hours in a 24 hour period (except as permitted by waiver).

A. Incorrect - the time is incorrect
B. Correct
C. Incorrect - both the time and approver are wrong
D. Incorrect – the approver is wrong

Technical Reference(s):.
1. APA-ZZ-00905, Limitations of Callaway Plant Staff Working Hours, Rev 21
2. CA-0161, Waiver of Work Hour Limits, Rev 0

References to be provided to applicants during examination: None

Learning Objective: T61.003A 6 – Normal Operations, LP #A-17, Objective A: DISCUSS the following as they pertain to APA-ZZ-00905, "Limitations of Callaway Plant Staff Working Hours":
1. The maximum permitted working hours in a given period
2. Required break periods
3. Minimum day off requirements
4. Difference between work hour limits during Normal Operation and Outages.
5. What time is excluded from calculating work hours
6. Waivers, including process for submitting and approving
7. Exceptions

Question Source: Bank # _x – no bank id – 2014 SRO Retake Audit Q#20_____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(1)

Comments:
SRO Only due to 43.1 - Conditions and limitations in the facility license as the work hour limitation and its program are apart of the facility license

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Conduct of Operations	**Group #**	Generic		
	K/A #	G 2.1.40		
	Importance Rating	3.9		
Knowledge of refueling administrative requirements.				

Question # 95

The Plant is in MODE 6 preparing for the start of Core Alterations, when it becomes necessary to perform Gate Valve Bypass Operations in accordance with Section 5.10 of OTS-KE-00015, Fuel Transfer System.

The MINIMUM approval required for this operation is the Refueling SRO and ….

A. Reactor Engineer

B. Director Nuclear Operations

C. Another Senior Reactor Operator

D. Westinghouse Fuel Representative

Answer: C

Explanation:

A. Incorrect – Plausible since RE is responsible for tracking fuel moves, etc.
B. Incorrect –Plausible since this position is responsible for fuel handling operations
C. Correct OTS-KE-00015 step 3.0
D. Incorrect –Plausible since Westinghouse approval is needed for inserting certain fuel assemblies back into core

Technical Reference(s):
1. ETP-ZZ-00035, REFUELING PERFORMANCE Rev 043
2. OTS-KE-00015, FUEL TRANSFER SYSTEM Rev 029

References to be provided to applicants during examination: None

Learning Objective: T61.0110 Systems, LP #79 Fuel Handling Systems, Objective H: Describe the interlocks and protective features of the following:
1. New fuel elevator

2. Spent fuel bridge crane
3. Transfer system
4. Refueling machine gripper

Question Source: Bank # ___X L16666___
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(7)

Comments:

SRO Only due to 10 CFR 55.43.b(6 and 7):

(6) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

(7) Fuel handling facilities and procedures.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	3		
Equipment Control	**Group**	Generic		
	K/A	G2.2.12		
	Importance Rating	4.1		
Knowledge of surveillance procedures.				

Question # 96

A surveillance requires authorization to start. Worker Protection Assurance is NOT required but Control Room Notification (WPANC) is required.

(1) Per APA-ZZ-00340, Surviellance Program Adminstration, who is required to authorize the above surviellance to start?

And

(2) Per APA-ZZ-00340, when the surviellance is complete, the surviellance status must be changed to COMP (complete) within a MAXIMUM of ___(2)______.

A. (1) Work Group Supervisor
(2) the end of shift

B. (1) Work Group Supervisor
(2) 1 working day

C. (1) Shift Manager / Control Room Supervisor
(2) the end of shift

D. (1) Shift Manager / Control Room Supervisor
(2) 1 working day

Answer: D

Explanation:

Per Section 5.14 of APA-ZZ-00340 –
5.14.1. Authorization to Start
a. All surveillances require authorization to start.
b. If WPA is required the Operations SM/CRS electronically authorizes the Surveillance Job to start on the 'Permits/WPA/EOSL/FPIP' tab of the Job Detail window.
c. If WPA is NOT required BUT Control Room Notification is required, the SM/CRS electronically authorizes the Surveillance Job to start on the 'Permits/WPA/EOSL/FPIP' tab of the Job Detail window.

d. If WPA is NOT required and Control Room Notification is NOT required, authorization to start is granted by the Work Group Supervisor by changing the job status to 'INPR' (In Progress).

Per 5.17. Processing Surveillance Packages from In Progress (INPR) to Final Closure (CLSD),

5.17.1. Work Group Supervisor changes Job status from FLDC to COMP (Complete) within one (1) working day of completing the job.

By the End of shift is plausible as there are several administrative requirement at the end of shift such as electronically signing of the job etc.

A: Incorrect – See above explanation – both are wrong
B: Incorrect – See above explanation – the authorization is incorrect
C: Incorrect – See above explanation – the time is incorrect
D: Correct -

Technical Reference(s):
1. APA-ZZ-00340, Surveillance Program Administration, Rev 44

References to be provided to applicants during examination: None

Learning Objective: T61.003A, Normal Operations, LP#A-14, B. DESCRIBE the following as they pertain to APA-ZZ-00340, Surveillance Program Administration:

1. Purpose and Scope
2. Discuss the responsibilities of Shift Manager/Control Room Supervisor.
3. Describe the Operating Authority for a Surveillance.
4. Describe the activities required for an Unsatisfactory or Partially Satisfactory Surveillance.
5. Define the Surveillance Frequency Control Program and where the Surveillance Frequency is maintained.

Question Source: Bank # ______
Modified Bank # ______
New ___X____

Question History: Last NRC Exam ___N/A_________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(1)

Comments:

SRO ONLY due to the condition and limitations in the facility license – The control and execution of the Surveillance Program is a condition of facility license

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	3		
Equipment Control	**Group**	Generic		
	K/A	G2.2.22		
	Importance Rating	4.7		
Knowledge of limiting conditions for operations and safety limits.				

Question # 97

The Plant is currently in MODE 4.

At 0400 today, it is discovered that a required routine 12-hour surveillance was last performed at 1200 on the previous day.

What describes the required action in response to the failure to perform the surveillance?

A. The surveillance requirements are satisfied if the surveillance is completed by 0600 per T/S SR 3.0.2.

B. The surveillance may be delayed, without declaring the LCO not met, for a maximum of 12 hours per T/S SR 3.0.3.

C. The surveillance may be delayed, without declaring the LCO not met, for a maximum of 24 hours per T/S SR 3.0.3.

D. The LCO must be declared not met and the surveillance completed within the specified frequency per T/S SR 3.0.3.

Answer: C

Explanation SR 3.0.0 - If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

A. Incorrect – see above
B. Incorrect see above
C. Correct – see above
D. Incorrect – see above

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Technical Reference(s):
1. Tech Specs SR 3.0.3

References to be provided to applicants during examination: None

Learning Objective: None

Question Source: Bank # __23431__
Modified Bank # ______
New _______

Question History: Last NRC Exam ___N/A___

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(2)

Comments:

SRO only due to 10 CFR55.43(b)(2) - Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	3		
Radiation Control	**Group #**	Generic		
	K/A #	G2.3.11		
	Importance Rating	4.3		
Ability to control radiation releases				

Question # 98

(1) A Containment Purge must be initiated within ____(1)_____ after the release permit is generated.

And

(2) The Shift Manager may authorize stopping and restarting a containment purge, without terminating the release permit, as long as the release permit is restated within a MAXIMUM of ______(2)_______.

A. (1) 12 hours
   (2) 75 minutes

B. (1) 12 hours
   (2) 120 minutes

C. (1) 24 hours
   (2) 75 minutes

D. (1) 24 hours
   (2) 120 minutes

Answer: B

Explanation:

Step 4.9 Containment Purge or Vent release is initiated within **12 hours after the release permit** is generated and within 24 hours after the collection of release samples from containment.

Step 4.11, Shift Manager may authorize stopping and restarting a containment purge, without terminating the release permit, as long as the release is restarted within two hours of the stop time.

A. Incorrect – See above explanation
B. Correct

C. Incorrect – See above explanation
D. Incorrect – See above explanation

Technical Reference(s):
1. HTP-ZZ-02012, Gaseous Radwaste Release Permit (Containment), Rev 52 steps 4.9 and 4.11

References to be provided to applicants during examination: None

Learning Objective: N/A

Question Source: Bank # ______
Modified Bank # ____X – no bank id – 2014 SRO Retake Audit Q#22______
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

10 CFR 55.43(b)(4)

Comments:

SRO only as it has to do with the approval process of release permits and times associated with these permits. Per the procedure, it is the Shift manager authorization / responsibility stopping and restarting a permit without terminating it and therefore a SRO ONLY Question / Topic.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Emergency Procedures / Plan	Group #	Generic		
	K/A #	G2.4.25		
	Importance Rating	3.7		
Knowledge of fire protection procedures				

Question # 99

Reactor Power is 100%.

Maintenance has requested control room activation of an FPIP for hot work in Room 1301 of the Auxiliary Building which is anticipated to cause excessive smoke in the area.

Room 1301 is protected by a smoke detector actuated pre-action sprinkler system.

(1) Per APA-ZZ-00742, who is responsible to ensure the pre-action sprinkler system is disabled when required to prevent inadvertent actuation?

And

(2) What department will perform the hourly fire watches associated with disabling the sprinkler system?

A. (1) Maintenance
(2) Security

B. (1) Maintenance
(2) Operations

C. (1) Control Room Supervisor
(2) Security

D. (1) Control Room Supervisor
(2) Operations

Answer: C

Explanation:

Per APA-ZZ-00742 – section 3.2 for Control Room responsibilities:

> *3.2.1. Responsible when notified of hot work, that HALON systems and pre-action sprinkler systems are disabled when required to prevent inadvertent actuation.*

Per APA-ZZ-00701 step 4.4.1 Control Room, COORDINATE with the following, as necessary, to establish necessary compensatory actions or measures:

- *Security to ensure hourly or continuous fire watches are posted*

Operations was plausible as they are the department that activates the impairments per APA-ZZ-00701 step 4.4.5. Maintenance is responsible for performing the hot work (welding / grinding/ etc.) making it a plausible but not responsible for ensuring it is disabled per APA-ZZ-00742.

A. Incorrect – wrong department
B. Incorrect – both are wrong
C. Correct – See above explanation
D. Incorrect – wrong department

Technical Reference(s):
1. APA-ZZ-00701, Control of Fire Protection Impairments, Rev 24
2. APA-ZZ-00742, Control of Ignition Sources, Rev 30

References to be provided to applicants during examination: None

Learning Objective: T61.0003A, Normal Operations, LP #30, FPIP, HAZMAT and Injuries, Objective D & B:
A. In accordance with APA-ZZ-00701; Control of Fire Protection Impairments
1. DEFINE Fire Protection Impairment
2. DEFINE Fire Protection Impairment Permit
3. DESCRIBE the Shift Manager/Control Room Supervisor/Shift Technical Advisor/Shift Engineer responsibilities.

D: In accordance with APA-ZZ-00742, Control of Ignition Sources
1. DESCRIBE the purpose and scope.
2. DEFINE:
    a. Combustible material
    b. Hot Work
    c. Hot Work Fire Watch
    d. Hot Work Permit Exempt Area

Question Source: Bank # ___X___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2017 ILT NRC Exam Question #98________

Question Cognitive Level:
Memory or Fundamental Knowledge __X___

Comprehension or Analysis _____

10 CFR Part 55 Content:

10 CFR 55.43(b)(1)

Comments:

SRO Only due to Conditions and limitation in the facility license 10 CFR 55.43(b)(1) specifically "Administration of fire protection program requirements such as compensatory actions associated with inoperable sprinkler systems, fire doors, etc."

This is within the job scope of a CRS/SM as shown in section 3 of APA-ZZ-00742 specifically step 3.2.1. "Responsible when notified of hot work, that HALON systems and pre-action sprinkler systems are disabled when required to prevent inadvertent actuation."

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier	3		
Emergency Procedures / Plan	**Group**	Generic		
	K/A	G2.4.21		
	Importance Rating	4.6		
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.				

Question # 100

The Incident Assessor is monitoring Critical Safety Function Status Trees (CSFST). A YELLOW path has been identified. The crew is performing the appropriate Functional Restoration Procedure (FRP).

(1) If the status of a different path changes to ORANGE, the CRS will transition to the ORANGE Path FRP ________.

And

(2) The CSFST shall now be monitored ________.

A. (1) immediately
(2) continuously

B. (1) immediately
(2) every 10 to 20 minutes

C. (1) after completion of the YELLOW path FRP
(2) continuously

D. (1) after completion of the YELLOW path FRP
(2) every 10 to 20 minutes

Answer: A

Explanation:
Per ODP-ZZ-00025, step 4.24.9.b.1 "When CSFST are applicable and after verifying that no RED condition exists, the Control Room staff is expected to stop the procedure in progress and implement the required FRP when a ORANGE condition arises. "

*Per ODP-ZZ-00025, step 4.24.10.a states "If a Red or Orange condition is encountered, the CSFST shall be monitored **continuously".** The distractor of 10-20 minutes is from step 4.24.10.b "When no condition more urgent than Yellow exists, the monitoring frequency should be*

every 10 to 20 minutes, unless some significant change in plant status occurs."

A. Correct
B. Incorrect – the monitoring requirement is wrong
C. Incorrect – the FRP transition is incorrect
D. Incorrect - both are wrong

Technical Reference(s):
1. CSF-1, Critical Safety Function Status Trees, Rev 13
2. ODP-ZZ-00025, EOP/OTO User's Guide, Rev 34

References to be provided to applicants during examination: None

Learning Objective:T61.003 D, Emergency Operations, LP #D-01, ERG Introduction and user's guide, Objective: AA. DESCRIBE the General Procedural Guidance provided by ODP-ZZ-00025, EOP/OTO User's Guide.

J. List the critical safety functions in order of priority and explain bases for this prioritization.

L. Explain operator responses during status tree monitoring for each of the following:

1. Extreme challenge is diagnosed
2. Severe challenge is diagnosed
3. Not satisfied condition is diagnosed

Question Source: Bank # _X___
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2014-1_Q#24_

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

10 CFR Part 55 Content:

10 CFR 55 43(b)(1)

Comments:

SRO Only due to Conditions and limitations of the facility license. The Emergency Plan is part of the facility license. Furthermore, ODP-ZZ-00001, Conduct of Operations, step 3.13.1 states "The IA position may be filled by an STA or SRO qualified individual." and step 3.13.3 directs the IA to monitor the CSF following a reactor trip or safety injection. To summarize, the incident assessor position is filled by a SRO and therefore a SRO Only function / topic.