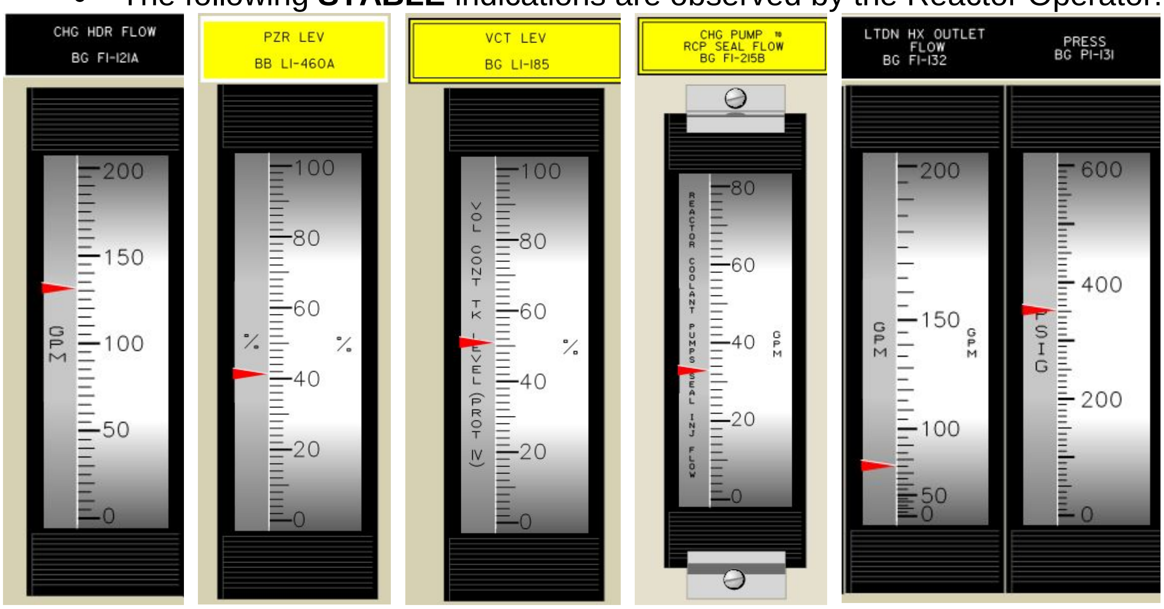
Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	1		
009 Small Break LOCA	**Group #**	1		
	K/A #	EA2.13		
	Importance Rating	3.6		
Ability to determine or interpret the following as they apply to a small break LOCA: Charging pump flow indication.				

Question #1

(REFERENCE PROVIDED)

Given the following plant conditions:

- Reactor power is 100%.
- An RCS Leak has been identified and the crew is performing OTO-BB-00003, RCS Excessive Leakage.
- Leak Isolation has **NOT** been successful.
- The following **STABLE** indications are observed by the Reactor Operator:



(1) In accordance with OTO-BB-00003, what action will be directed by the CRS?

And

(2) What is the **HIGHEST** EAL declaration that will be made?

A. (1) Commence Reactor Shutdown per OTO-MA-00008, Rapid Load Reduction
(2) Unusual Event

B. (1) Trip the Reactor and enter E-0, Reactor Trip or Safety Injection

(2) Unusual Event

C. (1) Commence Reactor Shutdown per OTO-MA-00008, Rapid Load Reduction
(2) Alert

D. (1) Trip the Reactor and enter E-0, Reactor Trip or Safety Injection
(2) Alert

Answer: A

Explanation: The leak rate calculation can be determined by reading the Charging header Flow (130 gpm), Letdown Flow (75 gpm), and the assumed normal RCP seal leakoff is 3 gpm per RCP. VCT and PZR level indication are provided to indicate that a Reactor Trip are not required due to low level in either. With the following calculation the leak rate is determined:
Charging Flow – Letdown Flow – RCP Seal Leak off = Leakage Rate
130-75-12=43 gpm

A. Correct – Based on OTO-BB-00003 Step 9 continuous action, the leak rate is LESS than 50 gpm, therefore a reactor trip is NOT required and the crew will perform Step 25 to commence a reactor shutdown in accordance with OTO-MA-00008, due to leakage in excess of T/S 3.4.13 RCS Operational Leakage exceeding the Unidentified Leakage Limit of 1 gpm. The leak rate is greater than 10 gpm Unidentified leakage and is therefore above the threshold for declaring an Unusual Even per SU6.1.
B. Incorrect – This is the correct EAL determination, however the leak rate is not greater than 50 gpm, PZR level is stable and VCT level is stable. A low PZR level, VCT level, or High leak rate reactor trip is NOT required. The correct action is to shutdown the reactor in a controlled manner.
C. Incorrect – Based on OTO-BB-00003 Step 9 continuous action, the leak rate is LESS than 50 gpm, therefore a reactor trip is NOT required and the crew will perform Step 25 to commence a reactor shutdown in accordance with OTO-MA-00008, due to leakage in excess of T/S 3.4.13 RCS Operational Leakage exceeding the Unidentified Leakage Limit of 1 gpm. This EAL determination is incorrect due to leak rate not exceeding 120 gpm. It is plausible because this is a compromise of the RCS barrier as indicated by the RCS leak
D. Incorrect – The leak rate is not greater than 50 gpm, PZR level is stable and VCT level is stable. A low PZR level, VCT level, or High leak rate reactor trip is NOT required. The correct action is to shutdown the reactor in a controlled manner. This EAL determination is incorrect due to leak rate not exceeding 120 gpm. It is plausible because this is a compromise of the RCS barrier as indicated by the RCS leak

Technical Reference(s):
1. OTO-BB-00003, RCS Excessive Leakage, Rev 22,
2. T/S 3.4.13 RCS Operational Leakage,
3. EIP-ZZ-00101 Addendum 1, Emergency Action Level Classification Matrix, Rev 3

References to be provided to applicants during examination:
1. EIP-ZZ-00101 Addendum 1, Emergency Action Level Classification Matrix, Rev 3

Learning Objective: T61.003B, LP-B-12 OTO-BB-00003, Obj. D. Given a set of plant conditions or parameters indicating excessive Reactor Coolant Leakage, 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:

55.43.5

SRO Only due CFR: 43.5 for EAL determination which is an SRO only function

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
000022 Loss of RX Coolant Makeup /2	Group #	1		
	K/A #	G2.4.45		
	Importance Rating	4.3		
Ability to prioritize and interpret the significance of each annunciator or alarm.				

Question # 2

Callaway is operating at 100% power when a transient occurs.

The following Annunciators are LIT:

- 32C PZR LO LEV DEV
- 38A LTDN REGEN HX TEMP HI
- 41A SEAL INJ TO RCP FLOW LO
- 41F NCP FLOW HILO

The following indications are observed by the Reactor Operator:

- Charging header flow is 130 gpm and stable.
- Pressurizer Level is 48% and stable.
- VCT level is 44% and slowly lowering.
- DRW Sump level is rising.

(1) Which of the following describes the event in progress?

And

(2) What action is required to mitigate this condition?

A. (1) RCP Seal Injection Header Rupture
(2) Perform OTN-BG-00001, Addendum 4, Operation of CVCS Letdown

B. (1) RCP Seal Injection Header Rupture
(2) Perform OTO-BB-00003, Attachment C, Auxiliary Building Leak Search

C. (1) Loss of air to BG FCV-124, NCP Flow Control Valve
(2) Perform OTO-BG-00001, Attachment H, Establishing Excess Letdown

D. (1) Loss of air to BG FCV-124, NCP Flow Control Valve
(2) Perform OTO-KA-00001, Attachment H, Air Operated Valves Outside Containment

Answer: B

Explanation:

The indications represent a RCP Seal Injection Header Leak is present – High NCP charging flow, low seal injection flow, and VCT level lowering with 75 gpm letdown in service. When Air is lost to BG FCV-124, the valve will Fail OPEN which would cause a high NCP flow, making this a plausible distractor, but would not cause RCP seal injection flow low annunciator. The values of the leak were chosen such that the leak rate, if calculated, would be less than 50 gpm. A leak rate of > 50 gpm would require a transition to E-0 making none of the choices correct. Also PZR level NOT stable or rising would cause a transition to E-0 because of step #2 of OTO-BB-00003, which is a continuous action step. Therefore, the question stem indicates that PZR level is stable.

For the procedure selection, with containment conditions normal (no information is provided that there are abnormal containment conditions, the RNO column of step 8 of OTO-BB-00003 would NOT be performed and would continue on in the procedure. The RNO column of step 13 applies (based on DRW level rising) and the CRS will direct Attachment C, Auxiliary Building Leak Search.

OTN-BG-00001, Addendum 4, Operation of CVCS Letdown is the normal operation procedure for CVCS for such activities as placing and removing letdown from service. This Addendum is not directed from OTO-BB-00003 but is directed from OTO-BG-00001.

If the candidate misdiagnoses the plant conditions and believes that a PZR level control malfunction is occurring (either due to a failed instrument or a FCV failure), then entry into OTO-BG-00001 is plausible and direction to isolate letdown and then establish excess letdown in step 7 RNO is plausible. This diagnosis is plausible as PZR level low out of band and a higher charging flow are given in the stem. This is wrong as none of this would explain why there is a seal injection flow low annunciator LIT. If the candidate believes that a loss of air to BG FCV 124 causing the valve to fail closed, then a loss of charging would have occurred and letdown would have isolated on low PZR level and after manual control of BG FCV 124 is taken in step 3, step 7 RNO would be applicable and hence why this choice is plausible.

OTO-KA-00001, Loss of Instrument Air, Attachment H will locate and isolate air to specific valves outside of containment. This would be the correct subsequent path, following OTO-BG-00001, to restore pressurizer level due to the loss of instrument air to BG FCV 124. However, as explained above, this is not the correct diagnosis of the event in progress but is plausible for a loss of air to the valve.

A. Incorrect. Wrong procedure selection
B. Correct.
C. Incorrect Both are wrong
D. Incorrect. Both are wrong

Technical Reference(s):.

1. OTO-BB-00003, RCS Excessive Leakage, Rev 22
2. OTO-BG-00001, Pressurizer Level Control Malfunction, Rev 20
3. OTO-KA-00001, Partial or Total Loss of Instrument Air, Rev 22
4. OTN-BG-00001, Chemical and Volume Control System, Rev 54
5. OTN-BG-00001, Addendum 4, Operation of CVCS Letdown, Rev 19

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

References to be provided to applicants during examination: None

Learning Objective: T61.003B6, Abnormal Operations, LP-B-12, Obj. D. Given a set of plant conditions or parameters indicating excessive Reactor Coolant leakage, 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: CFR 43.5

Comments:

SRO Only because this question involves Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. 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

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
000027 Pressurizer Pressure Control System Malfunction / 3	Group #	1		
	K/A #	AA2.11		
	Importance Rating	4.0		
Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: RCS Pressure				

Question # 3

Given the following plant conditions:

- Reactor power is 100%.
- The controlling Pressurizer Pressure Channel has just failed HIGH.

(1) What will actual RCS Pressure do immediately after the failure?

And

(2) The Low Pressurizer Pressure trip instrumentation provides protection to …..?

A. (1) rise
(2) prevent violating the DNBR Limit

B. (1) rise
(2) ensure that the allowable heat generation rate of the fuel is not exceeded

C. (1) lower
(2) prevent violating the DNBR Limit

D. (1) lower
(2) ensure that the allowable heat generation rate of the fuel is not exceeded

Answer: C

Explanation:

With a failed High pressure controlling channel, a signal will be processed to open the PZR spray valves in order to lower pressure. Therefore immediately after the failure, RCS pressure will lower.

The reason for the low pressurizer pressure trip is listed in the Technical Specification Bases of 3.3.1 RTS Instrumentation. Per Technical Specification bases of 3.3.1 function 8a– Pressurizer

Pressure Low; "Low trip Function ensures that protection is provided against violating the DNBR limit due to low Pressure". (Tech Spec bases page B3.3.1-21)

Per Technical Specification bases of 3.3.1 function 7; the distractor of "ensure that the allowable heat generation rate of the fuel is not exceeded" is the bases of overpower delta T trip function. Overpower delta T trip is T.S. 3.3.1 function 7. RCS Pressure is not an input into the overpower delta T trip setpoint determination. (Tech Spec bases page B3.3.1-19)

A. Incorrect – pressure would lower not rise
B. Incorrect – both are wrong
C. Correct
D. Incorrect – the bases reason is wrong.

Technical Reference(s):.
1. Technical Specification 3.3.1, RTS Instrumentation, and its bases
2. OTO-BB-00006, Pressurizer Pressure Control Malfunction, Rev 19

References to be provided to applicants during examination: None

Learning Objective:

T61.0110 6, Systems, LP #9, Reactor Coolant System, Objective B:
DESCRIBE the purpose and operation of the following RCS components to include interlocks, controller operations and power supply:

4. Pressurizer (Pzr)

Question Source: Bank # ______
Modified Bank # __X L7332____
New _______

Question History: Last NRC Exam ___N/.A_________

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

10 CFR Part 55 Content:

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?
- Knowledge of TS bases that is required to analyze TS required actions and terminology.

Yes SRO-only question

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
000056 Loss of Off-site Power / 6	Group #	1		
	K/A #	G2.4.41		
	Importance Rating	4.6		
Knowledge of the emergency action level thresholds and classifications				

Question # 4

(REFERENCE PROVIDED)

Given the following plant conditions:

- Reactor power is 100%.
- The B EDG, NE02, is tagged out for maintenance. The expected return is 24 hours from now.
- At 0800, a Loss of Off-Site Power occurs.
- At 0805, the transmission supervisor reports that Off-Site power should be restored to Callaway at 1400 the same day.

(1) Which of the following describes the **HIGHEST** Emergency Plan Action Level that applies to this situation?

And

(2) What is the **LATEST** notification time to the state and local agencies associated with this event?

A. (1) Unusual Event
(2) 0830

B. (1) Alert
(2) 0830

C. (1) Unusual Event
(2) 0915

D. (1) Alert
(2) 0915

Answer: B

Explanation:

Given the current plant status with a loss of offsite power, NB01 will be the only bus with power (supplied from NE01). This meets the ALERT criteria for SA1 "AC power capability buses reduced to a single power source for greater than or equal to 15 minutes such that any additional single failure would result in a station blackout". Therefore an ALERT declaration of SA1.1 is correct. The unusual event is plausible if the candidate does not correctly process the loss of NB02 due to the DG not being available and determines an EAL SU1.1, an Unusual Event, is the only emergency classification threshold reached due to the loss of off-site power.

Per APA-ZZ-00520, Attachment 1, Notification of state and local agencies is a 15 minute report upon the declaration of any emergency classification. Since the SRO has 15 minutes to declare the event, the latest the EAL declaration can occur is 0815. Then the SRO has an additional 15 minutes to notify state and local agencies which makes 0830 correct. The distractor of 0915 is plausible if the candidate incorrectly recalls that state and local agencies are notified within one hour (which is the notification time requirement to the NRC).

A. Incorrect – the EAL is wrong
B. Correct
C. Incorrect – Both are wrong
D. Incorrect – the notification time is wrong.

Technical Reference(s):.
1. EIP-ZZ-00101 Addendum 1, EAL Classification Level, Revision 3
2. APAZZ-00520, Reporting Requirements and Responsibilities, Rev 43 Attachment 1

References to be provided to applicants during examination:
1. EIP-ZZ-00101 Addendum 1, EAL Classification Level, Revision 3

Learning Objective:

T61.0110 Systems, LP #69, Event Review and Reportability Objective B:

PERFORM the following as they pertain to APA-ZZ-00520, REPORTING REQUIREMENTS AND RESPONSIBILITIES:

2. DISCUSS the incidents reportable in the following time frames:
   a. 15 minutes
   c. Immediate (1 hour)

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:

(CFR: 43.5)

SRO Only due to 43.1 - Conditions and limitations in the facility license for notification of outside agencies

SRO Only due CFR: 43.5 for EAL determination which is an SRO only function

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
057 Loss of Vital AC Inst. Bus	Group #	1		
	K/A #	AA2.14		
	Importance Rating	3.6		
Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: That substitute power sources have come on line on a loss of initial ac				

Question # 5

Given the following plant conditions:

- Reactor power is 100%
- Annun 26B, NN12 INV TRBL/XFR, alarms
- Computer Point NNU0003A, 1E, INV NN12 XFER TO ALT SPLY, is in alarm

The Secondary Operations Technician reports the following local indications:

- NN02 voltage is 120 VAC.
- P201, Inverter Supplying Load, light is NOT lit,
- P202, Bypass Source Supplying Load, light is LIT

Instrument Bus NN02 is powered by the ________(1)________ and the Instrument Bus NN02 is _________(2)_________.

	(1)	(2)
A.	Alternate power source via the Static Transfer Switch	Operable
B.	XNN06 Instrument Transformer using the sliding link breakers	Inoperable
C.	Alternate power source via the Static Transfer Switch	Inoperable
D.	XNN06 Instrument Transformer using the sliding link breakers	Operable

Answer: A

Explanation: The Static Transfer Switch automatically transfers to the alternate power source and illuminates the "Bypass Source Supplying Load" red light (P202) when the power from the inverter is lost. The XNN06 Transformer must be manually aligned to

supply power to NN02. Per TS Bases 3.8.7 and 3.8.9, the inverter is inoperable in the conditions listed but the NN bus is operable.

A. Correct per information above
B. Incorrect. not powered from XNN06 with condition given and NN02 is considered operable
C. Incorrect. NN02 is considered operable
D. Incorrect. not powered from XNN06

Technical Reference(s):.
1. Tech Spec Bases 3.8.7, and 3.8.9
2. OTS-NN-00012, NN12 INVERTER OUTAGE, Rev 23
3. OTN-NN-00002, 120V VITAL AC INSTRUMENT POWER-CLASS 1E (CHANNEL 2), Rev 6

References to be provided to applicants during examination: None

Learning Objective: T61.0110, Systems, LP-06, SAFEGUARDS POWER - NB/NG/NK/NN, Objective G, EXPLAIN the Technical Specifications and bases for the Safeguards Power System.

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:

55.43(b)2

Comments:

SRO based on knowledge of Tech Spec bases is required to determine that NN02 is Operable for the condition given.

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	1		
000077 Generator Voltage and Electric Grid Disturbances / 6	**Group #**	1		
	K/A #	G2.2.40		
	Importance Rating	4.7		
Ability to apply Technical Specifications for a system.				

Question # 6

(REFERENCE PROVIDED)

Given the following plant conditions:

- Reactor power is 100%.
- The A EDG, NE01, was tagged out for lube oil system maintenance 48 hours ago.
- An Electrical Grid Disturbance is in progress.
- The Transmission Operations Supervisor reports that a Category 8 Alarm is received. The Predicted Contingency Voltage is 325 kV.

In order to remain in compliance with the Technical Specifications, the reactor must be in MODE 3 within _______ hours?

A. 7

B. 14

C. 18

D. 30

Answer: A

Explanation:
Per Attachment 3 of OSP-NE-00003, "If the Predicted Voltage is outside of the required voltage range, the SM/CRS shall declare the offsite circuits INOPERABLE." Per Attachment 5 of OSP-NE-00003, the Contingency Analysis Computer Calculated Operability Limit is in the 372.6 – 329.8 kV. Therefore with a predicted analysis point less than 329.8 kV, the SRO candidate should declare both offsite circuits inop.

Based on the given plant conditions, Tech Spec 3.8.1 Conditions B, C, D, H are not met. The shortest time duration would be for 3.8.1. H which directs you to enter T.S. 3.0.3 immediately which requires Mode 3 within 7 hours.

30 hours is plausible if the candidate does not process that both offsite circuits are inop and proceeds with only Condition B (one DG inop) not met and then progresses to 3.8.1 G to be in mode 3 within an additional 6 hours. There are 24 hours left on the original time clock for condition B, therefore 24 + 6 = 30 hours. The candidate could also arrive at the 30 hour distractor by only applying 3.8.1C which has a 24 hour completion time. This time added with the 6 hours of 3.8.1 G would also yield a 30 hour time limit.

If the candidate applies 3.8.1 D (1 DG and 1 offsite circuit inop) as limiting (incorrectly), there is a 12 hour completion time before 3.8.1 G is applied. This would give the candidate a calculated time of 18 hours. (12 hours of D + 6 hours of G).

If the candidate applies the note in 3.8.1 condition D to enter LCO 3.8.9 condition A to restore within 8 hours and then proceeds to applies 3.8.9 Condition D to be in Mode 3 within 6 hours, the calculated time would be 14 hours. While this is a correct application it is not the most limiting time to be in Mode 3.

A. Correct
B. Incorrect – not the most limiting time applied 3.8.9 A then D
C. Incorrect – not the most limiting time applied 3.8.1 D then G
D. Incorrect – not the most limiting time applied 3.8.1 B then G (or C then G)

Technical Reference(s):
1. OSP-NE-00003, Technical Specifications Actions – A.C. Sources Rev 28
2. Technical Specification Section 3.8, Amendment #202

References to be provided to applicants during examination:
1. Technical Specification Section 3.8.1, AC Sources Operating
2. Technical Specification Section 3.8.9, Distribution Systems

Learning Objective:
T61.0110 Systems, LP #6 – Safeguards Power Objective G: EXPLAIN the Technical Specifications and bases for the Safeguards Power System.

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:

(CFR: 43.5)

SRO Only due to 43(b) #2 - Facility operating limitations in the technical specifications and their bases.

Additionally per Figure 1 Attachment 2 of ES-401,

- Can question be answered solely by knowing ≤ 1 hour TS/TRM Action? **Is No**
- Can question be answered solely by knowing the LCO/TRM information listed "above-the-line?" **is No**
- Can question be answered solely by knowing the TS Safety Limits? **Is 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) **is Yes which means this is an SRO Only question**

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
000036 Fuel Handling Accident / 8	Group #	2		
	K/A #	AA2.03		
	Importance Rating	4.2		
Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Magnitude of potential radioactive release				

Question # 7

Given the following plant conditions:

- The plant is in Mode 6.
- Core Alterations are in progress.
- While lowering a fuel assembly into its Spent Fuel Pool Storage rack location, a malfunction occurs causing the assembly to free fall the entire distance.
- Bubbles appear to be coming from the vicinity of the dropped assembly.
- Fuel Building Area Rad Monitor indications, SD RE-37 & 38, are rising rapidly.
- Fuel Building Atmosphere Monitor, GG RE-27, is in HI-HI alarm.
- Fuel Building Atmosphere Monitor, GG RE-28, is constant and NOT in alarm.

(1) What is the status of the Emergency Exhaust System?

And

(2) The reactor must be subcritical for a minimum of 72 hours before the movement of irradiated fuel from the RX vessel can occur ___________?

A. (1) ONLY one train actuated
(2) to ensure spent fuel pool boron concentration is > 2165 ppm

B. (1) BOTH trains actuated
(2) to ensure spent fuel pool boron concentration is > 2165 ppm

C. (1) ONLY one train actuated
(2) to minimize the potential release from a fuel handling accident

D. (1) BOTH trains actuated
(2) to minimize the potential release from a fuel handling accident

Answer: D

Explanation:
Per technical specification bases 3.3.8 background section "High gaseous radiation, monitored by two channels, provides a FBVIS. Both EES trains are initiated by high radiation detected by either channel. Each channel contains a gaseous monitor. High radiation detected by either monitor initiates fuel building isolation, starts the EES, and initiates a CRVIS." This information is also repeated in FSAR section 15.7.4.5 Radiological Consequences. Therefore BOTH trains actuated is correct.

Per OSP-SF-00003, step 6.5.2, the reactor is required to be subcritical for at least 72 hours prior to the start of core alterations. This is to ensure that iodine inventory low enough which minimizes the potential offsite dose due to a fuel handling accident.

Both OSP-SF-00003 and OTG-ZZ-00007 have steps prior to movement of Irradiated fuel assemblies in the fuel building to "ENSURE the Fuel Storage Pool boron concentration is greater than or equal to 2165 ppm." but this is not the reason why fuel moves are delayed until 72 hours after the reactor is subcritical. Plausible as this is a precaution, note, and step in the procedures and it does take time to raise and ensure boron concentration is >2165 ppm.

Specifically: OSP-SF-00003 section 5.8 additional action - core off-load, the Note prior to step 5.8.2 and OTG-ZZ-00007 step 6.8.6.

A. Incorrect – Both are wrong
B. Incorrect – the reason is wrong
C. Incorrect – both trains would actuate
D. Correct

Technical Reference(s):.
1. OTO-KE-00001, Fuel Handling Accident, Rev 14
2. Technical Specification 3.3.8, Emergency Exhaust System (EES) Actuation Instrumentation and its bases.
3. Technical Specification 3.7.13, Emergency Exhaust System (EES) and its bases.
4. FSAR Section 15.7.4, Fuel Handling Accidents, page 15.7-12
5. OSP-SF-00003, Pre Core Alteration Verifications, Rev 27 step 6.5.2 and attachment 4.
6. OTG-ZZ-00007, Refueling Preparation, Performance and Recovery, Rev 36

References to be provided to applicants during examination: None

Learning Objective:
T61.003E – Refueling Operations, LP #E-5, Objective I;
Describe the Purpose, Symptoms or Entry Conditions, and major action steps of OTO-KE-00001, "FUEL HANDLING ACCIDENT."

T61.0110 Systems, LP #39, Objective D;
LIST the signals that cause a Fuel Building. Ventilation Isolation Signal (FBVIS) and DESCRIBE the sequence of events that occur on a FBVIS.

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:

SRO only due to 43.6 - 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. [10 CFR 55.43(b)(6)] Specifically Administrative requirements associated with refueling activities. Additionally [10 CFR 55.43(b)(1)] applies - Conditions and limitations in the facility license because the 72 hours is listed as an assumption in the fuel handling accident analysis contained in the FSAR.

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
000076 High Reactor Coolant Activity / 9	Group #	2		
	K/A #	G2.4.11		
	Importance Rating	4.2		
Knowledge of abnormal condition procedures.				

Question # 8

(REFERENCE PROVIDED)

Given the following plant conditions:

- The plant is in Mode 4.
- SJ RE-01, CVCS Letdown Monitor, is in alarm and indicates 30 µCi/ml and slowly rising.
- PZR Level is 30% and constant.
- Charging flow is constant.
- Chemistry reports that Dose Equivalent I-131 is 75 µCi/gm.

(1) The Control Room Supervisor shall direct the crew to ____________?

And

(2) Which of the following describes the **HIGHEST** Emergency Plan Action Level that applies to this situation?

A. (1) isolate Letdown
(2) Unusual Event

B. (1) isolate Letdown
(2) Alert

C. (1) maximize Letdown flow through CVCS Letdown Mixed Bed Demineralizer
(2) Unusual Event

D. (1) maximize Letdown flow through CVCS Letdown Mixed Bed Demineralizer
(2) Alert

Answer: D

Explanation:
The entry conditions for OTO-BB-00005, High RCS activity, have been met for 2 reasons – Dose Equivalent I-131 and SJ-RE-01 in alarm. The first action of this procedure, i.e step 1, is to ***maximize flow through the CVCS Letdown Mixed Bed Demineralizer****. The distractor of isolate Letdown is plausible since the candidate may falsely believe the action is to isolate Letdown to reduce the spread of contamination through the CVCS system. This is wrong as it is the exact opposite of what OTO-BB-00005 requires the operator to do.*

Based on the plant conditions, an Unusual Event condition of Fuel clad degradation exists (dose Equivalent I-131 greater than 75 µCi/gm for greater than 48 hours). This would result in a SU5.1 declaration if it was the only threshold met. But an unusual event is not the highest threshold met ***as the condition for FA1.1 have been met due to a loss of the fuel clad barrier due to Radiation due to the CVCS letdown radiation monitor, SJ-RE-01, reading more than 25 µCi/ml.*** *Therefore the highest EAL is an Alert due to FA1.1, any Loss or any potential loss of either Fuel Clad or RCS.*

A. Incorrect – Both are wrong
B. Incorrect – The action is wrong
C. Incorrect – The EAL is incorrect
D. Correct

Technical Reference(s):.
1. EIP-ZZ-00101 Addendum 1, EAL Classification Level, Rev 3
2. OTO-BB-00005, RCS High Activity, Rev 14

References to be provided to applicants during examination:
1. EIP-ZZ-00101 Addendum 1, EAL Classification Level, Rev 3

Learning Objective:
T61.003 B Off Normal Operations – LP #B-14
- Objective C: DESCRIBE symptoms or entry conditions for OTO-BB-00005, RCS High Activity.
- Objective D Given a set of plant conditions or parameters indicating RCS High Activity, 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:

SRO Only due to 43.1 - Conditions and limitations in the facility license

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
W/E03 LOCA Cooldown - Depress. / 4	**Group #**	2		
	K/A #	EA2.1		
	Importance Rating	3.4		
Ability to determine and interpret the following as they apply to the (LOCA Cooldown and Depressurization): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.				

Question # 9

Given the following plant conditions:

- A LOCA has occurred.
- The crew is performing E-1, Loss of Reactor or Secondary Coolant.

The following parameters exist:

- o All SG pressures are 900 psig and slowly trending down.
- o All SG levels are 40% NR and stable.
- o PZR level is off-scale low.
- o RVLIS PUMPS OFF indication is 20%.
- o Containment Pressure is 23 psig and rising slowly.
- o RWST level is 69% and decreasing slowly.
- o RCS pressure is 750 psig and decreasing slowly.

Based on these indications, what procedure will the crew enter next?

A. ES-1.1,SI Termination

B. ES-1.2, Post LOCA Cooldown and Depressurization

C. ES-1.3, Transfer to Cold Leg Recirculation

D. E-2, Faulted Steam Generator Isolation

Answer: B

Explanation:

A. Incorrect Transition to ES-1.1 would occur if RCS pressure was stable or rising and PZR level was greater than 9%. The stem stated that RCS pressure was slowly lowering and PZR level was off scale low. (See E-1 Step 6)

B. Correct Transition to ES-1.2 is directed by E-1 Step 13 when RCS pressure is greater than 325 psig.
C. Incorrect Transition to ES-1.3 would occur if RWST level lowered to less than 36%. The conditions stated in the stem have RWST level at 69% and slowly lowering. (See E-1 Step 14)
D. Incorrect Transition to E-2 would occur if any SG pressure was lowering in an uncontrolled manner. The stem stated that SG pressure is slowly lowering. (see E-1 Step 2)

Technical Reference(s):.
E-1, Loss of Reactor or Secondary Coolant, Rev 17, Step 13
ES-1.2, Post LOCA Cooldown and Depressurization, Rev 14

References to be provided to applicants during examination: None

Learning Objective: Lesson plan D-10, ES-1.2 POST LOCA COOLDOWN AND DEPRESSURIZATION, Obj B, DESCRIBE the Symptoms and/or Entry conditions for ES-1.2, Post LOCA Cooldown and Depressurization.

Question Source: Bank # __L16447____
Modified Bank # ______
New _______

Question History: Last NRC Exam ___2007_________

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

10 CFR Part 55 Content:

(CFR: 43.5)

Comments:

SRO per criteria 5 “Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures”

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	1		
W/E10 Natural Circulation with Steam Void in Vessel with/without RVLIS / 4	Group #	2		
	K/A #	G2.4.46		
	Importance Rating	4.2		
Ability to verify that the alarms are consistent with the plant conditions.				

Question # 10

Given the following plant conditions:

- A Reactor trip occurred due to a loss of offsite power.
- Shortly after the trip, the BOP reports the following annunciators are LIT:
  - o 25A, NN01 INST BUS UV
  - o 57E, RVLIS PWR Failure
- The operating crew is performing ES-0.2, Natural Circulation Cooldown.
- RCS pressure is 1920 psig.
- The RCS cooldown and depressurization MUST be performed due to secondary systems water inventory concerns.
- It is suspected that a steam void has formed in the RX Vessel.

(1) Which of the following annunciators can be used to verify that a steam void has formed in the RX Vessel?

And

(2) The CRS will direct which of the following procedures?

A. (1) 32A, PZR Level High
(2) Transition to ES-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS)

B. (1) 32A, PZR Level High
(2) Transition to ES-0.4, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS)

C. (1) 33C, Pressurizer Pressure Low – Heaters On
(2) Transition to ES-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS)

D. (1) 33C, Pressurizer Pressure Low – Heaters On

(2) Transition to ES-0.4, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS)

Answer: A

Explanation:
'A' RVLIS is powered from NN01 so a loss of NN01 means that A RVLIS is not available. Therefore the 'A' train of RVLIS is inoperable but the B Train is operable.

And per the NOTE prior to step 13 in ES-0.2 that states:
"If at any time it is determined that a natural circulation cooldown and depressurization must be performed at a rate that may form a steam void in the vessel, one of the following procedures should be used:
ES-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS) or
ES-0.4, Natural Circulation Cooldown With Steam Void In Vessel (Without RVLIS)"

Therefore, ES-0.3 is correct based on plant conditions and the fact that one train of RVLIS is operable.

If A Steam void is suspected of forming in the vessel, this void will force water into the pressurizer and annunciate 32A, PZR level high. Pressurizer pressure would be going up not down as the PZR bubble would be squeezed by the incoming surge. This question is basically modeling the TMI accident with the exception of a failed open PZR PORV. With a LOCA in progress, it is plausible that a low PZR Pressure Alarm will be received. While there is no LOCA in this question, 33C is plausible if the student applies the TMI accident concept from memory without understanding the reason. Furthermore, step 13 of ES-0.2, RNO for part C directs using a PZR PORV as letdown would not be in service. Opening the PZR PORV would give the PZR Low alarm as pressure is relieved to the PRT. However, as explained above, the Note prior to step 13 would direct the operator to either ES-0.3 or ES-0.4 and the crew would not be performing step 13. Additionally, the PORV operation leading to a low PZR pressure alarm is plausible as certain steps in ES-0.2 direct use of a PZR PORV which would create a low PZR Pressure.

RCS pressure of 1920 psig indicates that the crew is at step 12 of ES-0.2.

A. Correct
B. Incorrect
C. Incorrect
D. Incorrect

Technical Reference(s):
1. ES-0.3, Natural Circulation Cooldown with Steam Void In Vessel (with RVLIS), Rev 12
2. ES-0.4, Natural Circulation Cooldown with Steam Void In Vessel (without RVLIS), Rev 11
3. EOP Addendum 1, Natural Circulation Verification, Rev 2
4. ES-0.2, Natural circulation Cooldown, Rev 11
5. The following list of Annunciator Response Procedures:
    a. OTA-RK-25A
    b. OTA-RK-32A
    c. OTA-RK-32D
    d. OTA-RK-56B
    e. OTA-RK-57C

f. OTA-RK-57D
g. OTA-RK-57E

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:

G. STATE and EXPLAIN the parameters which are evaluated, including their Criteria and Basis, to transition from the following procedures to other procedures:
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 _____
Comprehension or Analysis ___X__

10 CFR Part 55 Content:

(CFR: 43.5)

SRO Only due to 43.5. Specifically, per page ES-401 Page 20-21 which states:
"The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item, for example:

• Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures."

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
012 Reactor Protection	Group #	1		
	K/A #	G2.4.30		
	Importance Rating	4.1		
Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.				

Question # 11

Which of the following requires notification to the NRC Resident Inspector in accordance with ODP-ZZ-00001 Addendum 13, Shift Manager Communications?

A. Unplanned entry into an OTO procedure.

B. An employee's injury has been classified as a lost time away accident.

C. Scheduled Tech Spec outage on the 'A' CCP took 75% of the allowed out of service time.

D. Unplanned entry into Tech Spec 3.3.1, RTS Instrumentation, required action that has a 48 hour completion time.

Answer: D

Explanation:

A. Incorrect - Per ODP-ZZ-00001 Addendum 13, Attachment 1 page 2, notification of the NRC Resident Inspector is not for an unplanned OTO entry
B. Incorrect - Per ODP-ZZ-00001 Addendum 13, Attachment 1 page 4, notification of the NRC Resident Inspector is not for a lost time accident/injury
C. Incorrect - Per ODP-ZZ-00001 Addendum 13, Attachment 1 page 3, notification of the NRC Resident Inspector is not for a TS outage exceeding.
D. Correct - Per ODP-ZZ-00001 Addendum 13, Attachment 1 page 1, notification of the NRC Resident Inspector is required for an unplanned TS entry with less than 72 hour action statement. See Note 1 of the matrix "Entry into Tech. Spec. action statement with 72 hours or less completion time."

Technical Reference(s):
1. ODP-ZZ-00001 Addendum 13, Shift Manager Communications, Rev 17

References to be provided to applicants during examination: None

NRC Site-Specific Written Examination
Callaway Plant
Senior Reactor Operator

Learning Objective: LP-66 Operations Department-Code of Conduct Obj, B, EXPLAIN the following as they pertain to Operations Department Communications., 2. Addendum 13 of ODP-ZZ-00001, Shift Manager communications to emergency duty officer

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:

(CFR 43.1) SRO per criteria 1 due to the reporting requirements associated with the facility license

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
013 Engineered Safety Features Actuation	Group #	1		
	K/A #	A2.05		
	Importance Rating	4.2		

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

Loss of dc control power

Question # 12

Given the following plant conditions:

- Reactor Power is 100%.
- Annunciator 25B, NN11 INV TRBL/XFR, alarms.
- Annunciator 25C, NK01 TROUBLE, alarms.
- NK EI-I, 125V DC BUS NK01 VOLT indicates 0 volts.

(1) What is the status of the A train of EFSAS?

And

(2) To verify proper alignment of the Turbine Driven Auxiliary FeedWater system, the CRS will direct which of the following procedures?

A. (1) SB066X indications will be red
(2) OTN-AL-00001, Auxiliary Feedwater System

B. (1) SB066X indications will be red
(2) OTO-SA-00001, Attachment AH, AFAS/LSP Train A Verification

C. (1) SB066X indications will be white
(2) OTN-AL-00001, Auxiliary Feedwater System

D. (1) SB066X indications will be white
(2) OTO-SA-00001, Attachment AH, AFAS/LSP Train A Verification

Answer: B

Explanation: Per OTO-NK-00002 Attachment A Loss of power to NK01, "Loss of control power to ESFAS Cabinet SA036A results in loss of ESFAS Train A automatic and manual

actuation". Furthermore, "Loss of control power to ESFAS Train A Solid State Load Sequencer Panel NF039C results in loss of Train A load shed and sequencing capability." *Based on the initial plant conditions and a loss of NK01, which will cause all 4 FWIVs to close, a reactor trip will be required.*

OTO-SA-00001 Attachment AH is correct as a TDAFW actuation occurred on low low SG levels and due to the loss of indication of A Train EFSAS (SA066X) this procedure attachment is correct to verify A Train EFSAS.

OTN-AL-00001 is incorrect as it provides instruction for normal standby lineup and / or manual operation of the TDAFW pump. Section 5.1 directs Checklist 1, Auxiliary Feedwater Valve Alignment only provides for a normal valve alignment. These are plausible if the candidate believes that NO TDAFS exists or believes that this procedure provides direct for verifying alignment after an actuation signal.

A. Incorrect – wrong procedure selection
B. Correct
C. Incorrect – Both are incorrect
D. Incorrect – incorrect SA066X indications

Technical Reference(s):.
1. OTO-NK-00002, Loss of Vital 125 VDC Bus, Rev 13
2. OTO-SA-00001, ESFAS Verification and Restoration, Rev 39
3. OOA-SA-C066X, Engineered Safety Feature (ESF) Status Panel SA066X Alarm Information, Rev 14
4. E-0, Reactor Trip or Safety Injection, Rev 16
5. OTN-AL-00001, Auxiliary Feedwater System, Rev 33

References to be provided to applicants during examination: None

Learning Objective:

T61.003B 6, Off Normal Operations, LP #B50, OTO-NK-00002,

Objective E: ANALYZE OTO-NK-00002 and DETERMINE the conditions that would require a Reactor Trip/Turbine Trip

Objective C: Given a set of plant conditions or parameters indicating a Loss of Vital 125 VDC Bus, 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:

(CFR: 43.5)

SRO Only because this question involves Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. 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

Comments:

See below for a desktop simulator run of a loss of NK01 and its effect on SA066X (A Train EFSAS)

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
022 Containment Cooling	Group #	1		
	K/A #	G2.1.32		
	Importance Rating	4.0		
Ability to explain and apply system limits and precautions.				

Question # 13

(REFERENCE PROVIDED)

Given the following plant conditions:

- 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 plant operational restrictions due to 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 insturmentaion,
2. TS and Bases 3.6.6 Containment spray and cooling system,
3. OTN-GN-00001, Containment Cooling and CRDM Cooling, Rev 28

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 # ______
Modified Bank # __X L16440____
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

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**

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
063 DC Electrical Distribution	**Group #**	1		
	K/A #	A2.01		
	Importance Rating	3.2		

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

Question # 14

Given the following plant conditions:

- Reactor Power is 100%.
- The Crew is preparing to investigate a ground on NK01.
- When NKHS00001, GROUND TEST SWITCH, is placed in the test positions; a negative ground is indicated.
- Breaker NK0111, FDR BKR TO 7.5 KVA INVERTER NN11 will be the first breaker to be OPENED.

(1) IF the ground **IS** isolated after NK0111 is opened, the operator should expect the ground test voltmeter to indicate approximately ____________ when tested.

And

(2) The CRS will direct ground isolation in accordance with what procedure?

A. (1) 65 VDC
(2) OTO-NK-00001 Attachment A, Actions for NK01

B. (1) 130 VDC
(2) OTO-NK-00001 Attachment A, Actions for NK01

C. (1) 65 VDC
(2) OTN-NK-00001 Addendum 1, 125VDC Bus NK01 and Distribution System

D. (1) 130 VDC
(2) OTN-NK-00001 Addendum 1, 125VDC Bus NK01 and Distribution System

Answer: C

Explanation:

A. Incorrect. In accordance with OTN-NK-00001 Addendum 1, 125VDC Bus NK01 and Distribution System section 5.3, Ground Detector Operation, when the grounded circuit is opened, the Ground Meter indication for the affected bus will return to a nominal value (60 to 70 VDC). OTO-NK-00001, Failure of NK Battery Charger addresses the loss of the battery charger and the subsequent actions to be taken to ensure loads supplied by the bus NK01 via the battery are functioning properly. The Attachment A actions for NK01 will ensure all instruments are not selected for control or functioning properly. This attachment does not specifically address Ground issues on the bus.

B. Incorrect. If the candidate mistakenly believes that the Ground Meter Indication should indicate approximately the expected battery voltage (125VDC nominal) then they could assume that the grounded circuit is the lower reading circuit (65VDC). OTO-NK-00001, Failure of NK Battery Charger addresses the loss of the battery charger and the subsequent actions to be taken to ensure loads supplied by the bus NK01 via the battery are functioning properly. The Attachment A actions for NK01 will ensure all instruments are not selected for control or functioning properly. This attachment does not specifically address Ground issues on the bus.

C. Correct. In accordance with OTN-NK-00001 Addendum 1, 125VDC Bus NK01 and Distribution System, when the grounded circuit is opened, the Ground Meter indication for the affected bus will return to a nominal value (60 to 70 VDC). OTN-NK-00001 Addendum 1, 125VDC Bus NK01 and Distribution System section 5.4 directs actions to be taken to perform Breaker flashing for locating a ground on the bus.

D. Incorrect. If the candidate mistakenly believes that the Ground Meter Indication should indicate approximately the expected battery voltage (125VDC nominal) then they could assume that the grounded circuit is the lower reading circuit (65VDC). OTN-NK-00001 Addendum 1, 125VDC Bus NK01 and Distribution System section 5.4 directs actions to be taken to perform Breaker flashing for locating a ground on the bus.

Technical Reference(s):.

1. OTN-NK-00001 ADD 01, 125VDC BUS NK01 AND DISTRIBUTION SYSTEM, Rev 3;
2. OTO-NK-00001, Failure of NK Battery Charger, Rev 13

References to be provided to applicants during examination: None

Learning Objective: T61.0110.6, LP-06, Obj. M. EXPLAIN the precautions, limitations and bases for the following components/conditions associated with OTN-NK-00001, “Class 1E 125 VDC Electrical System”

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__

Note: this is a higher order question as the candidate is given a set of plant conditions and must predict what the correct reading would be when the malfunction is corrected. Furthermore, the candidate has to analyze the situation and determine that the plant is not in an abnormal situation and plant activities will be controlled with a normal operating procedure.

10 CFR Part 55 Content:

(CFR: 43.5)

Comments:

K/A Match: This question requires the operator to predict the expected indications based on the actions to be taken during the ground Breaker Flashing process. And Select the appropriate procedure which will be used to perform the ground locating process.

SRO Only: This question is SRO only based on the 43.5, selection of the appropriate plant procedure based on the indication of a ground on the 125VDC NK Bus to isolate the ground. The candidate must determine if the correct direction is located in the attachment to the abnormal operating procedure or in the addendum to the normal operating procedure.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
103 Containment	Group #	1		
	K/A #	G2.4.4		
	Importance Rating	4.7		
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.				

Question # 15

A major LOCA has occurred while operating at 100% reactor power. The crew is currently in E-1, Loss of Reactor Or Secondary Coolant, responding to the LOCA.

The Shift Technical Advisor (STA) reports the following containment parameters associated with the Containment Critical Safety Function:

- Containment Pressure — 25 psig and stable
- Containment Normal Sump Level — 98 inches and stable
- Containment Radiation — 3.4 rad/hour and stable

Which of the following actions should be taken by the Control Room Supervisor to respond to the containment conditions reported by the STA?

A. Go To FR-Z.1, Response to High Containment Pressure, due to an Orange Path

B. Go To FR-Z.1, Response to High Containment Pressure, due to a Yellow Path
OR
Continue in E-1

C. Go To FR-Z.2, Response to Containment Flooding, due to an Orange Path

D. Go To FR-Z.3, Response to High Containment Radiation Level, due to a Yellow Path
OR
Continue in E-1

Answer: D

Explanation:

A. Incorrect – For the conditions given there is NOT an ORANGE path for high containment pressure. Orange path requires containment pressure to be greater than 27 psig and NO containment spray running
B. Incorrect – For the conditions given there is NOT a YELLOW path for high containment pressure. Yellow path requires containment pressure to be greater than 27 psig with ONE containment spray pump running. For Yellow paths the CRS has the choice of continuing with the procedure and step in effect or going to the FR procedure for the affected CSF
C. Incorrect – For the conditions given there is NOT an ORANGE path for containment flooding. Orange path requires containment normal sump level to be greater than 106 inches.
D. Correct – For the condition given a YELLOW path exists for high containment radiation levels (greater than 3 rad/hour. For Yellow paths the CRS has the choice of continuing with the procedure and step in effect or going to the FR procedure for the affected CSF

Technical Reference(s):.CSF-1, Critical Safety function Status Trees, Rev 10 page 9

References to be provided to applicants during examination: None

Learning Objective: Lesson plan D-01, ERG Introduction and User's Guide, Objectives K Explain how challenges to critical safety functions are prioritized within each critical safety function and 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 # ______
Modified Bank # __R14980____
New _______

Question History: Last NRC Exam ____N/A________

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

10 CFR Part 55 Content:

(CFR: 43.5)

Comments:

SRO per criteria 5 "Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures."

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Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
016 Non-nuclear Instrumentation	Group #	2		
	K/A #	A2.01		
	Importance Rating	3.1		

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

Detector failure

Question # 16

Given the following plant conditions:

- The Callaway Plant is in MODE 4.
- RCS pressure is being controlled at 500 psig.
- All wide range Cold Leg temperatures are 270°F.
- Cold Overpressure Protection is in ARMED.
- Loop 1 Wide Range Cold Leg temperature sensor, TE413B, fails low.

Subsequently;

- The Crew has transitioned to OTO-BB-00010, Shutdown LOCA.
- The Reactor Operator reports that Pressurizer Level is lowering.

(1) Which of the following describes the plant response to this failure?

And

(2) The CRS will direct which action to mitigate this event?

A. (1) Only the B Train PORV, BB PCV 456A, will open.
(2) Restore SI Pumps to be capable of injection per OSP-EM-00002, Section 7.1 Restoring SI System.

B. (1) Only the B Train PORV, BB PCV 456A, will open.
(2) Restore SI Accumulators per OTN-EP-00001, Addendum 6, SI Accumulator Isolation and Restoration.

C. (1) Both PORVs, BB PCV 455A and BB PCV 456A, will open.
(2) Restore SI Pumps to be capable of injection per OSP-EM-00002, Section 7.1 Restoring SI System.

D. (1) Both PORVs, BB PCV 455A and BB PCV 456A, will open.
   (2) Restore SI Accumulators per OTN-EP-00001, Addendum 6, SI Accumulator Isolation and Restoration.

Answer: A

Explanation: *The is a common misconception that for this situation (with COMS Armed) that both PZR PORVs will open due to a failure of the Loop 1 WR* T_{cold}. *The inputs into the logic are as follows:*

	T hot	***T cold***	***WR Pressure***	***PORV***
A Train	***1,2***	***3,4***	***BB PT 405***	***BB PCV 455A***
B Train	***3,4***	***1,2***	***BB PT 406***	***BB PCV 456A***

Therefore, when Loop 1 T cold Fails Low ***ONLY BB PCV 456A will be affected.***

Per Annunciator 35B PORV open, step 3.3 – 'IF the PORVs should NOT be OPEN:

3.3.1. IF excessive RCS leakage is indicated, Go To OTO-BB-00003, Reactor Coolant System Excessive Leakage." The report that PZR level is lowering means that the crew will NOT stay in OTO-BB-00003 and transition to OTO-BB-000010, Shutdown LOCA. This is given in the stem as the entry conditions for OTO-BB-00010, are RO knowledge and not being tested here. OTO-BB-00010 step 10 directs that SI and CCP system be realigned as they have been placed in a lineup for COMS. OSP-EM-00002 section 7.1 is correct as it would restore SI system for injection. OTN-EP-00001 is plausible as it would also met the same strategy of recovering an ECCS system for injection into the vessel but is NOT directed from OTO-BB-00010 and therefore wrong. SI accumulators are pressurized to ~650psig and with RCS pressure at 500 psig they could inject.

A. Correct
B. Incorrect – wrong procedure
C. Incorrect – wrong # of PORV opening
D. Incorrect – Both reasons are wrong

Technical Reference(s):.
1. OTA-RK-00018, Addendum 35B, PORV Open, Rev 0
2. OTO-BB-00003, RCS Excessive Leakage, Rev 22
3. OTG-ZZ-00006, Plant Cooldown Hot Standby To Cold Shutdown, Rev 72
4. OTO-BB-00010, Shutdown LOCA, Rev 4
5. E-0, Reactor Trip or Safety Injection, Rev 16
6. OTN-EP-00001, Accumulator Safety Injection System, Rev 26
7. OSP-EM-00002, Rendering SI pumps Incapable of Injection, Rev 22
8. M-22BB-01 Rev 31, Mechanical Draw for RCS with TE and Coms circuit inputs.

References to be provided to applicants during examination: None

Learning Objectives:

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T61.0110.6, systems, LP #30, Reactor Instrumentation - SC,

Objective B: DESCRIBE the purpose, characteristics and operation (normal and abnormal) of the following Reactor Instrumentation and Pressurizer Pressure/Level Control System components:
1. Wide Range (WR) Temperature Instruments

Objective C: LIST the outputs of the WR and NR Temperature Instruments, including auctioneering circuits.

T61.003B – 6, Off Normal Operations, LP #64, OTO-BB-00010,

Objective C DESCRIBE Continuous Action Step(s) including the required Response Not Obtained actions.

Objective D Given a set of plant conditions or parameter indicating a Shutdown LOCA, ANALYZE the correct procedure(s) to be utilized and the required actions to stabilize the plant.

Question Source: Bank # ______
Modified Bank # ___X L5857_- Bank Question is at Ro level __
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 55.43.5)

SRO Only because this question involves Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. 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

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
029 Containment Purge	Group #	2		
	K/A #	G2.1.40		
	Importance Rating	3.9		
Knowledge of refueling administrative requirements.				

Question # 17

Given the following plant conditions:

- Callaway is in Mode 6 with Core Off load in progress.
- Shutdown Purge is in service.
- GT-RE-0022 and GT-RE-0033, CTMT Purge EXH Detectors, are in Bypass.
- Preparations are being made to open the Equipment Hatch.

1) With the Equipment Hatch Open and core alterations in progress, what is the required Administrative Control?

And

2) In the event of a fuel handling accident, the Containment Purge Isolation System will be actuated _____________ containment closure is completed.

A. (1) Ensure dedicated individuals are available to close the equipment hatch.
(2) before

B. (1) Ensure a dedicated individual is available to restore GT-RE-0022 and GT-RE-0033 to Operate.
(2) before

C. (1) Ensure dedicated individuals are available to close the equipment hatch.
(2) after

D. (1) Ensure a dedicated individual is available to restore GT-RE-0022 and GT-RE-0033 to Operate.
(2) after

Answer: C

Explanation:

A. Incorrect.
B. Incorrect.
C. Correct. In Mode 6 during core alterations, TS 3.9.4 requires that containment closure be obtainable through the use of administrative controls which require: 1) Appropriate Personnel are notified, 2) Specify individuals designated and readily available to Close all open containment penetrations, 3) All obstructions that would prevent rapid closure of the penetration can be quickly removed to allow closure of the penetration. OTN-GT-00001, Containment Purge System, precaution and limitation 3.5 requires that during core alterations containment purge exhaust must meet the requirements of T/S 3.9.4 Administrative Controls. The basis of T/S 3.9.4 requires the closure of containment in a specified sequence in the event of a fuel handling accident, 1) Manually Actuate Control Room Vent Isolation, 2) Close Containment Hatches (Emergency Air Lock, and Personnel Airlock), and 3) following closure of the personnel airlock and emergency air lock, Manually Initiate Containment Purge Isolation System (CPIS).
D. Incorrect.

Technical Reference(s):.
1. T/S 3.9.4 bases
2. OTN-GT-00001, Containment Purge System, Rev 30

References to be provided to applicants during examination: None.

Learning Objective: T61.0110. 6 Systems, LP #40, Containment Ventilation,

Objective M: DESCRIBE function and operation of the following containment purge system components.

1. Mini Purge Supply Air Unit
2. Shutdown Purge Supply Air Unit
3. Containment Purge Filter Absorber Unit
4. Mini Purge Exhaust Fan
5. Shutdown Purge Exhaust Fan

Objective O: STATE the Limiting Conditions for Operation (LCO) AND Bases for the following Containment Ventilation System related Technical Specifications (T/S):

16. T/S 3.9.4

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:

(CFR: 43.6)

Comments:

SRO Only due to 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.
[10 CFR 55.43(b)(6)]

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	2		
015 Nuclear Instrumentation	Group #	2		
	K/A #	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

Question # 18

(REFERENCE PROVIDED)

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 I&C has corrected the cause of the failure.
  - 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) The CRS will direct which of the following procedures?

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, Rev 15
3. OSP-SE-00003, Quadrant Power Tilt Ration Calculation, Rev 21
4. OSP-SF-00002, Control Rod Partial Movement, Rev 22
5. OTA-RK-00022, ADD 81B, Rod at Bottom, Rev 2
6. ESP-ZZ-00004, Flux and Thermocouple Mapping, Rev 15
7. ESP-ZZ-00006, Incore/Excore Calibration, Rev 32

References to be provided to applicants during examination:
1. Curve Book, Figure 8-7, RCS LOOP with Control Rods and Excore Neutron Detector Locations, Rev. 000

Learning Objective:
T61.GFES, Reactor Operational Physics, LP #44, Objective 22: Explain reactor response to a control rod insertion.

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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 # ______
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:

(55.43.5)

SRO Only because this question involves Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. 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

Comments:

Examination Outline Cross-reference:	**Level**	**SRO**		**Rev 0**
	Tier #	3		
Conduct of Operations	**Group #**			
	K/A #	G2.1.40		
	Importance Rating	3.9		
Knowledge of refueling administrative requirements.				

Question # 19

The Callaway 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.

Which of the following identifies the **MINIMUM** requirements for approval of this operation?

A. The Refueling SRO ONLY

B. The Reactor Engineer ONLY

C. The Refueling SRO and another SRO

D. The Reactor Engineer and another SRO

Answer: C

Explanation: Per OTS-KE-00015, step 5.10.1 states "REQUEST permission from the Refueling SRO and a second SRO for Gate Valve Bypass Operations." Therefore the minimum requirements are the refueling SRO and another SRO.

A. Incorrect - another SRO is required
B. Incorrect - Reactor Engineers are not licensed SROs and 2 SROs are needed at a minimum
C. Correct
D. Incorrect - Reactor Engineers are not licensed SROs and 2 SROs are needed at a minimum

Technical Reference(s):.
1. ETP-ZZ-00035, Refueling Performance, Rev 37
2. OTS-KE-00013, Refueling Machine, Rev 31
3. OTS-KE-00015, Fuel Transfer System, Rev 25

References to be provided to applicants during examination: None

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Learning Objective:
T61.003E – Refueling Operations, LP #E-5, Objective H;
Describe the interlocks and protective features of the following:
3. Transfer system

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:

(CFR: 43.7)

SRO only due to 43.7 - Fuel handling facilities and procedures.

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Conduct of Operations	Group #			
	K/A #	G2.1.37		
	Importance Rating	4.6		
Knowledge of procedures, guidelines, or limitations associated with reactivity management.				

Question # 20

Operations management is **REQUIRED** to designate an additional SRO to fulfill the Reactivity Management SRO (RMSRO) position for which of the following situations?

A. Planned load reduction from 100% to 97%.

B. A rapid load reduction due to an emergent plant event.

C. RCS Dilution for Low Power Physics Testing post refuel.

D. The plant is in Mode 6 with portions of the RCS being filled from the RWST.

Answer: C

Explanation:
Per ODP-ZZ-00001 Addendum 10, step 2.1.5 states that "Designate a Reactivity Management SRO (RMSRO) to direct Reactor Operations whenever reactor power is changed by more than 5 % in one direction. This includes:

- ***Reactor startup***
- ***Lowering power in MODES 1 and 2***
- ***Raising power in MODES 1 and 2***
- ***Withdrawal of Shutdown Banks***
- ***Diluting for Physics Testing post refuel."***

Therefore, RCS Dilution for Low Power Physics Testing post refuel is correct. Lowering power from 100% to 97% does not meet the power changed by more than 5% criteria and is therefore incorrect.
ODP-ZZ-00001, step 3.6.1 states 'The CRS normally performs the responsibilities of the RMSRO during steady state and emergent plant conditions. The CRS may also perform the RMSRO function during reactor startup.' Therefore, since this is an emergent power reduction operations management is NOT required to designate an additional SRO to fulfill the RMSRO position.
Step 2.1.12 defines what activities are potential reactivity manipulations due to dilution activities which would require additional briefing and peer check but these don't require a RMSRO. Specifically "Filled portions of the RCS and the Refueling Pool (RFP) that have direct access to the Reactor Vessel in mode 6 below 2000 ppm" was selected but is incorrect. Since RWST boron concentration is directed per UFSAR 16.1.2.5 for Mode 6 to

be greater than 2350 ppm there is no way this evolution would result in going below 2000 ppm.

A. Incorrect – 5%power change rule
B. Incorrect – an emergent power reduction
C. Correct
D. Incorrect – Listed as a potential reactivity manipulation due to dilution in step 2.1.12 of addeddum 10 but does not require a RMSRO.

Technical Reference(s):.
1. APA-ZZ-01300, Reactivity Management Program, Rev 21
2. ODP-ZZ-00001, Addendum 10, Reactivity Management, Rev 16
3. ODP-ZZ-00001 Conduct of Operations, Rev 91, Section 3.6.2

References to be provided to applicants during examination: None

Learning Objective:
T61.0110 Systems, LP#66, Operations Department Code of Conduct, Objective A;
EXPLAIN the following as applied in ODP-ZZ-00001, Operations Dept. – Code of Conduct:
4.Reactivity management

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:

(CFR: 43.6)

SRO Only due to 43.6 - 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.

Specifically, the administrative requirements associated with low power physics testing apllies making this an SRO Only question.

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Equipment Control	Group #			
	K/A #	G2.2.6		
	Importance Rating	3.6		
Knowledge of the process for making changes to procedures.				

Question # 21

Given the following plant conditions:

- The plant is in Mode 3.
- Engineering has requested that the A SI pump be started with the discharge valve throttled to 75% open to determine starting current.
- A Special Test procedure has been developed and approved.
- The Director of Operations has determined that a major revision to the Special Test Procedure is required.

The Shift Manager may approve the test procedure change ____________

A. without any restrictions.

B. ONLY after licensing concurrence is obtained.

C. ONLY with concurrence from another licensed SRO.

D. ONLY after a written 10CFR50.59 review has been approved.

Answer: D

Explanation:

A. Incorrect, special test required a 50.59 review
B. Incorrect, special test required a 50.59 review
C. Incorrect, if this was a temporary change concurrence from another licensed SRO is needed.
D. Correct, a special test requires written 50.59 safety evaluation

Technical Reference(s):

1. APA-ZZ-00101, Processing Procedures, Manual, and Desktop Instructions, Rev 68
2. APA-ZZ-00143, 10CFR50.59 Review, Rev 15

References to be provided to applicants during examination: None

Learning Objective: A-14, F. STATE the following as they pertain to APA-ZZ-00101 – Processing Procedures, Manuals, and Desktop Instructions:

1. The Purpose and Scope
2. When Administrative Correction Revisions may be performed
3. When Temporary Changes may be performed
4. SRO role in Temporary Change process
5. Reviews required for Major/Minor Revisions and New Procedures

Question Source: Bank # _X_L16466____
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2007________

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

10 CFR Part 55 Content:

(CFR: 43.3)

Comments:

SRO per criteria 3 “Facility licensee procedures required to obtain authority for design and operating changes in the facility.
• 10 CFR 50.59 screening and evaluation processes.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Equipment Control	Group #			
	K/A #	G2.2.22		
	Importance Rating	4.7		
Knowledge of limiting conditions for operations and safety limits.				

Question # 22

Given the following plant conditions:

- At 0800, it was determined that a surveillance requirement which was due at midnight the previous shift was not performed.
- The surveillance requirement frequency is once per 72 hours.
- The surveillance was performed satisfactorily 80 hours ago.

What is the **LATEST** time, from 0800, in which the surveillance requirement must be completed satisfactorily to be within its <u>Specified Frequency</u>.

A. 10 hours

B. 18 hours

C. 24 hours

D. 72 hours

Answer: A

Explanation:

This question involves the application of SR 3.0.2 and SR 3.0.3. First it is necessary to calculate the 25% grace period that is specified in SR 3.0.2. 72 hours x 0.25 = 18 hours. Since the surveillance is still within its grace period, the remainder of the grace period will be used to perform the surveillance within its specified frequency. The time remaining on the grace period is 10 hours (18 hours – (80-72 hours)) as 8 hours of the grace period has elapsed by 0800. Therefore, 10 hours is the Latest time in which the SR may be completed sat and the specified frequency be met.

A. Correct – see above
B. Incorrect – plausible if the candidate calculates the grace period correctly but does not apply the correct starting time of the grace period i.e. assumes the grace period starts at 0800 instead of when the normal 72 hours was up (i.e 8 hours ago).

C. Incorrect – plausible if the candidate does not understand that SR 3.0.3 is used for situations when the specified frequency in NOT met. SR 3.0.3 provides 2 time periods: "from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater" 24 hours is plausible is the candidate thinks it is 24 maximum or believes it says "whichever is less".
D. Incorrect – plausible if the candidate does not understand that SR 3.0.3 is used for situations when the specified frequency in NOT met. SR 3.0.3 provides 2 time periods "from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater" 72 hours is plausible if the candidate remembers SR 3.0.3 states the maximum of the specified frequency.

Technical Reference(s):.
1. Technical Specification Section SR4.0.1 thru SR4.0.4
2. Technical Specification Section 1.4, Frequency

References to be provided to applicants during examination: None

Learning Objective: T61.0110 6 Systems, LP #77, Introduction to Technical Specifications, Objective G EXPLAIN and APPLY the LCO/SR applicability section of Technical Specifications

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:

(CFR: 43.2)

This is SRO only because in involves the application of Application of generic Limiting Condition for Operation (LCO) requirements (LCO 3.0.1 thru 3.0.7; SR 3.0.1 thru 3.0.4) per B. Facility operating limitations in the TS and their bases i.e.10 CFR 55.43(b)(2)

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Radiation Control	Group #			
	K/A #	G2.3.13		
	Importance Rating	3.8		

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question # 23

Access to a locked high radiation area is required for Category 1: Life Saving.

(1) The recommended dose limit (Deep Dose Equivalent) for Category 1: Life Saving is ________?

And

(2) Who can authorize this exposure?

A. (1) 10 rem
(2) Emergency Coordinator

B. (1) 10 rem
(2) Manager, Radiation Protection

C. (1) 100 rem
(2) Emergency Coordinator

D. (1) 100 rem
(2) Manager, Radiation Protection

Answer: C

Explanation:

Per HDP-ZZ-01450, Attachment 1, Category 1 Life Saving recommended dose limit is 100 rem DDE. Also per HDP-ZZ-01450 and CA 0276, 4 people can authorize this dose exposure in excess of the limits:

- *Senior Vice President Generation and Chief Nuclear Officer*
- *Vice President Nuclear Operations*
- *Emergency Coordinator*
- *Recovery Manager*

None of which are the Manager of Radiation Protection (RPM). The RPM is plausible based on their responsibilities during normal operations which is outlined in APA-ZZ-01000.

A. Incorrect – wrong dose limit
B. Incorrect – both are wrong
C. Correct
D. Incorrect – wrong person

Technical Reference(s):.
1. HDP-ZZ-01450, Authorization to Exceed Federal Occupational Dose, Rev 11
2. CA 0276, Authorization to Exceed Federal Occupational Radiation Dose Limits, Rev 12/11/13
3. APA-ZZ-01000, Callaway Energy Center Radiation Protection Program, Rev 40

References to be provided to applicants during examination: None

Learning Objective:

T61.0110 Systems , LP #75 – ALARA and RB Entry, Objective I

1. IDENTIFY who can authorize dose exposure in excess of 10CFR20.1201 dose limits.
2. DISCUSS the limits for plant emergencies and the selection criteria associated with these limits

Question Source: Bank # __L16627____
Modified Bank # ______
New _______

Question History: Last NRC Exam _____2013_______

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

10 CFR Part 55 Content:

SRO Only due to Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)].

Emergency dose authorization is a SRO function / responsibility as the Emergency coordinator position is filled by a SRO.

Comments:

K/A match as this question tests about the knowledge of radiological safety procedures pertaining to licensed operator duties specifically access to locked high-radiation areas. Fulfilling the Emergency Coordinator position is a SRO licensed operator duty.

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Emergency Procedures / Plan	**Group #**			
	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 # 24

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 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*

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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 10
2. ODP-ZZ-00025, EOP/OTO User's Guide, Section 4.26, Rev 26

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 # ______
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:

(CFR: 43.1) – 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.

Comments:

Examination Outline Cross-reference:	Level	SRO		Rev 0
	Tier #	3		
Emergency Procedures / Plan	Group #			
	K/A #	G2.4.38		
	Importance Rating	4.4		
Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.				

Question # 25

An Emergency has been declared.

You are the Emergency Coordinator.

Which of the following responsibilities may you delegate?

A. Classifying and declaring emergencies.

B. Requesting the formation of emergency teams.

C. Authorizing personnel exposure in excess of 10CFR20 limits.

D. Decision making for implementing strategies identified in the Severe Accident Management Guidelines.

Answer: B

Explanation:

Per EIP-ZZ-00102, step 3.1 states which responsibilities may or may not be delegated specifically:

3.1. Emergency Coordinator (EC) is responsible for implementing this procedure and directing emergency response as follows: [Ref: 6.2.6]

*3.1.1. The following Emergency Coordinator responsibilities **may NOT be delegated**:*

- *Classifying and declaring emergencies*
- *Authorizing personnel exposure in excess of 10CFR20 limits*
- *Decision making for implementing strategies identified in the Severe Accident Management Guidelines*

*3.1.2. The following actions **may be delegated** by the Emergency Coordinator:*

- *Directing operations of Emergency Response Organization*
- *Requesting the formation of emergency teams*
- *Initiating implementation of onsite protective actions*
- *Ensuring that Emergency Response Organization are kept up-to-date on emergency conditions*

- *Ensuring that site-wide announcements are made on the plant Public Address (PA) system*

A. Incorrect
B. Correct
C. Incorrect
D. Incorrect

Technical Reference(s):.

1. EIP-ZZ-00240, Technical Support Center Operations, Rev 42

References to be provided to applicants during examination: None

Learning Objective:
None

Note: There are no objectives in System's LP #76, EIPs, for EIP-ZZ-00102.

Question Source: Bank # __X L 14399___
Modified Bank # ______
New _______

Question History: Last NRC Exam ____2005________

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

10 CFR Part 55 Content:

(CFR: 43.1) Conditions and limitations in the facility license.

The Emergency plan is part of the facility license and, additionally, the Emergency Coordinator position filled by SRO's making this a SRO responsibility and therefore a SRO Only question.

Comments: