FINAL

QUESTIONS REPORT
for VCS_2009_NRC_SRO_EXAM_DATABASE_-_AS_GIVEN

1. 001 AA2.03 002

Given the following plant conditions:

- Following a Refueling Outage, the crew is performing a power ascension from 80% power.
- Control rods began moving OUT at 72 SPM.
- Control rods were placed in MAN, and rod motion is stopped.
- Control bank 'D' rods stepped out 14 steps.
- Tavg is 580°F and continues to rise.

Which ONE (1) of the following statements identifies the **FIRST** condition that will occur and the action required by AOP-403.3, *Continuous Control Rod Motion*?

A. A reduction in the OTΔT trip setpoint will occur;

Adjust Control Rods to maintain Tavg within 5°F of Tref.

B.✓ A reduction in the OTΔT trip setpoint will occur;

Adjust Control Rods to maintain Tavg within 1°F of Tref.

C. The Technical Specification 3.2.5 DNB Parameter for Tavg will be exceeded;

Adjust Control Rods to maintain Tavg within 5°F of Tref.

D. The Technical Specification 3.2.5 DNB Parameter for Tavg will be exceeded;

Adjust Control Rods to maintain Tavg within 1°F of Tref.

A. Plausible because the 1st part is correct - the setpoint for OTΔT will be penalized as soon as Tavg starts increasing. 2nd part is plausible because Control Rods will be used to adjust Tavg in AOP-403.3, *Continuous Control Rod Motion,* and because other control rod-related AOPs (403.5, 403.6) require Tavg to be maintained within 5°F of Tref.

Incorrect because Tavg must be maintained within *1°F* of Tavg in AOP-403.3.

B. CORRECT. The OTΔT setpoint is calculated to prevent exceeding DNB and the setpoint will be penalized as soon as Tavg starts increasing (see IC-6, pages 16-21). AOP-403.3 Step 5 requires Control Rods to be adjusted to maintain Tavg within 1°F of Tref.

C. Plausible because, at BOL (following RF outage), the DNB Parameter limit of 589.2°F for Tavg may be exceeded. 2nd part is plausible because Control Rods will be used to adjust Tavg in AOP-403.3, *Continuous Control Rod Motion,* and because other control rod-related AOPs (403.5, 403.6) require Tavg to be maintained within 5°F of Tref.

Incorrect because the OTΔT setpoint is will be penalized <u>as soon as</u> Tavg starts increasing. Tavg must increase to 589.2°F from the given value of 580°F. Also incorrect because Tavg must be maintained within *1°F* of Tavg in AOP-403.3.

D. Plausible because the 2nd part is correct. Also plausible because, at BOL (following RF outage), the DNB Parameter limit of 589.2°F for Tavg may be exceeded. A small MTC at BOL would mean that Tavg would have to significantly change to compensate for the given rod movement (14 steps). Also plausible because, for situations where the OTΔT setpoint is penalized by large changes in *ΔI*; there could potentially be enough time lag for Tavg to exceed the DNB Parameter limit before the OTΔT setpoint is actually penalized by ΔI.

Incorrect because the OTΔT setpoint is will be penalized <u>as soon as</u> Tavg starts increasing. Tavg must increase to 589.2°F from the given value of 580°F.

K/A - 001 AA2.03:

Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place.

Tier: 1
Group: 2

Importance Rating: SRO 4.8

Technical Reference:

- IC-6, pp 16-21

Proposed references to be provided to applicants during examination:

None

Learning Objective: IC-6-16

Question History: NEW

10 CFR Part 55 Content: 43(b)(2)

K/A Match:

Not an exact match to the K/A because any action taken as a result of a failure of an automatic action would be an Immediate Operator Action; therefore, would not be an SRO Only question. Question was written to the impact that a Continuous Rod Withdrawal has on the automatic calculation of a protection setpoints and the bases for that calculation.

SRO Only Justification:

SRO Only because the question tests knowledge of a detailed Step in AOP-403.3, which is NOT and Immediate Operator Action.

2. 001 G2.4.34 003

Given the following plant conditions:

- 100% power
- AOP-600.1, *Control Room Evacuation*, has been entered.
- Because of the immediate nature of the evacuation, NO operator actions were taken prior to leaving the Control Room.
- The NROATC reports to the CRS that he is unable to trip the Reactor from the MG set room and is proceeding to the local switchgear.
- The CRS has lined up both CREPs and is determining how long to Emergency Borate.
- Reactor power is NOT decreasing.

Based on these conditions, Emergency Boration from the CREPs must continue until ________.

A. the Control Room can be re-entered

B. RCPs have been run for GREATER THAN 15 minutes

C. natural circulation has been established

D.✓ Either NI-32A or NI-33 indicates 1000 cps or less

A. Plausible because the thought process may be that entering the CR would allow the crew to attempt shutdown from the Reactor Control Station of the MCB.

Incorrect because the table in Step 12 of AOP-600.1 requires boration until S/D, regardless of how much time has elapsed.

B. Plausible because RCPs must be run until Emerg Boration is complete. Also plausible because, in AOP-600.1, 15 minutes is the time frame that the Reactor must be tripped to energize NI-33.

Incorrect because the table in Step 12 of AOP-600.1 requires boration until S/D, regardless of how much time has elapsed.

C. Plausible because natural circulation would allow for mixing if RCPs had been previously secured.

Incorrect because the table in Step 12 of AOP-600.1 requires boration until S/D, regardless of how much time has elapsed.

D. CORRECT. Since Reactor power is NOT decreasing, according to the table in Step 12 of AOP-600.1, EMERG BORATE required until S/D on EITHER NI-32A or NI-33.

K/A - 001 G2.4.34:

(Control Rod Drive) Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Tier: 2
Group: 2

Importance Rating: SRO 4.1

Technical Reference:

- AOP-600.1

Proposed references to be provided to applicants during examination:

None

Learning Objective: AOP-600.1-2258

Question History:

NEW

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

The K/A is matched because the operator must demonstrate knowledge of the RO tasks (Emerg Boration)), performed outside the Control Room, associated with the CRDS (Reactor NOT tripped); during an emergency (CR evacuation), and within the context of resultant operational effects.

SRO Only Justification:

The question is SRO-Only because it requires the operator to recall detailed procedural steps.

3. 003 AA2.03 002

Given the following plant conditions:

- 85% power.
- Power Range channel N-41 is failed and has been removed from service.
- One Control Bank 'C' rod bottom light energized and Control Bank 'D' moved out 10 steps.
- A Quadrant Power Tilt Ratio (QPTR) indicates a QPTR of 1.06.

Which ONE(1) of the following completes the statement below?

Reactor Engineering <u>must</u> confirm the QPTR using ____________ within ___ hours in accordance with Technical Specification 3.2.4.

A. BEACON ONLY;

TWO (2)

B. BEACON ONLY;

TWELVE (12)

C. BEACON or In-Core moveable detectors;

TWO (2)

D.✓ BEACON or In-Core moveable detectors;

TWELVE (12)

A. 1st part plausible because BEACON *is* a software system which obtains IPCS data and uses it to update a real time analytical core model. This core model may be used for Tech Spec surveillances, and is also referred to as the Power Distribution Monitoring System (PDMS). (Ref. STP 212.001 *Core Power Distribution Measurement* rev. 12). Also plausible because TWO (2) hours is the amount of time that T.S. 3.2.4, Action a. allows to reduce QPTR within limits.

Incorrect because BEACON is <u>not</u> the <u>only</u> method used per T.S. 3/4.2.4. Also incorrect because Surveillance Requirement 4.2.4.2 requires that the QPTR be determined within the limit when above 75 percent rate thermal power with one Power Range Channel inoperable at least once per ***12*** hours by using the PDMS (aka BEACON) <u>**or**</u> Movable Incore Detectors to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR.

B. Plausible because the 2nd part is correct. Surveillance Requirement 4.2.4.2 stipulates that the QPTR be determined within the limit when above 75 percent rate thermal power with one Power Range Channel inoperable at least once per

12 hours by using the PDMS **or** Movable Incore Detectors to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR. 1st part is plausible because BEACON *is* a software system which obtains IPCS data and uses it to update a real time analytical core model. This core model may be used for Tech Spec surveillances, and is also referred to as the Power Distribution Monitoring System (PDMS). (Ref. STP 212.001 *Core Power Distribution Measurement* rev. 12)

1st part incorrect because BEACON is not the only method used per T.S. 3/4.2.4 Surveillance Requirement 4.2.4.2. Surveillance Requirement 4.2.4.2 stipulates that the QPTR be determined within the limit when above 75 percent rate thermal power with one Power Range Channel inoperable at least once per ***12*** hours by using the PDMS **or** Movable Incore Detectors to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR.

C. Plausible because the 1st part is correct. T.S. 3/4.2.4 surveillance requirement 4.2.4.2 requires that the QPTR shall be determined within the limit when above 75 percent rate thermal power with one Power Range Channel inoperable at least once per 12 hours by using the PDMS **or** Movable Incore Detectors to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR. Also plausible because TWO (2) hours is the amount of time that T.S. 3.2.4, Action a. allows to reduce QPTR within limits.

Surveillance Requirement 4.2.4.2 stipulates that the QPTR be determined within the limit when above 75 percent rate thermal power with one Power Range Channel inoperable at least once per ***12*** hours, not 2.

D. CORRECT. T.S. 3/4.2.4 Surveillance Requirement 4.2.4.2 requires that the QPTR shall be determined within the limit when above 75 percent rate thermal power with one Power Range Channel inoperable at least once per 12 hours by using the PDMS **or** Movable Incore Detectors to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR.

K/A - 003 AA2.03:

Ability to determine and interpret the following as they apply to the Dropped Control Rod: Dropped rod, using in-core/ex-core instrumentation in-core or loop temperature measurements.

Tier: 1
Group: 2

Importance Rating: SRO 3.8

Technical Reference:

- T.S. 3/4.2.4
- T.S. 6.9.1.11 pp 6-16, 16a
- STP-212.001, p 3 of 17

Proposed references to be provided to applicants during examination: None

Learning Objective: SB-4-18

Question History: NEW

10 CFR Part 55 Content: 43(b)(2)

K/A Match:

Matches the K/A in that the question tests the methodology for conducting a T.S. surveillance after a dropped rod.

SRO Only Justification:

SRO Only in that it tests T.S. Actions that are greater than ONE (1) hour.

4. 008 G2.4.6 003

Given the following plant conditions:

- 100% power
- Following a plant transient, a PZR Code Safety Valve lifts and remains stuck in the **FULL OPEN** position.
- An automatic Reactor Trip and Safety Injection occur.
- The crew has entered EOP-2.1, *Post-LOCA Cooldown and Depressurization*, and taken the proper actions.
- The crew has just verified that SG Narrow Range levels are >40%.
- The NROATC notes the following conditions:
    - RCS subcooling is 20°F.
    - The PZR is water-solid.
    - PZR pressure is 1000 psig.

IAW EOP-2.1, which ONE (1) of the following identifies what actions are required to be taken next AND the basis for this decision?

A.✓ Initiate an RCS cooldown first

Then start RCP 'A' to mix the reactor coolant to a uniform temperature.

B. Initiate an RCS cooldown first

Then start RCP 'A' to ensure that PZR Spray is available for RCS pressure control.

C. Start RCP 'A' to mix the reactor coolant to a uniform temperature first.

Then initiate an RCS cooldown

D. Start RCP 'A' to ensure that PZR Spray is available for RCS pressure control first.

Then initiate an RCS cooldown

A. CORRECT. The plant has experienced a Small Break LOCA and will enter EOP-1.0, Reactor Trip/Safety Injection Actuation, transition to EOP-2.0, Loss of Reactor or Secondary Coolant, and then to EOP-2.1, Post-LOCA Cooldown and Depressurization. At Step 5, the crew will initiate an RCS cooldown to Cold Shutdown. At Step 8, they will check for adequate subcooling (>30°F). At step 9 of EOP-2.1 the crew will evaluate whether or not an RCP should be started. Since a PZR Steam Space Accident has occurred, PZR level will be off-scale high within minutes of the event. The ES-1.2 WOG Background Document for Step 12 (HES12BG.doc, HP-Rev 2, 4/30/05, p94) states that forced circulation is the preferred mode of operation to allow for normal RCS cooldown and provide PZR spray. A knowledge item associated with this step (p95) states that even if plant specific procedures require that a steam bubble be present in the PZR (which they do [SOP-101, Precaution 2.a.6.d, Rev 26]), if an RCS leak path is certain (which is the case in the stated conditions), RCP restart should be permitted since the leak ensures that there will NOT be a significant surge when the RCP is started. This is restated on page 44, in a section analyzing a Steam Space Small Break LOCA. This section also indicates that the RCP is not "needed for normal spray capability but served to mix the RCS coolant to nearly uniform temperature." EOP-2.1, Step 9 (Rev 13), reflects the requirements of the Background Document. When checking to see if an RCP should be started, the crew is first directed to check that all RCPs are secured (which they are). Then, the crew is directed to check RCS Subcooling > 30°F and PZR level > 30% [50%]. When these are satisfied (as is the case in the established conditions), the operator is directed to (1) establish normal conditions for starting an RCP, refer to SOP-101, and (2) start RCP 'A'. (RCP 'A' is the preferred RCP since it is attached to the surge line and will provide the most effective PZR spray control)

B. Plausible because pressure control is part of the normal bases for starting an RCP. Also plausible because the 1st part is correct.

Incorrect because spray valves will not be effective in pressure control with the PZR water-solid.

C. Plausible because normally, an RCP should be started to ensure that PZR Spray is available for RCS pressure control and to mix reactor coolant to uniform temperatures. Also plausible because RCP 'A' should normally be atarted to provide the most effective spray since it is attached to the surge line.

Incorrect because subcooling is insufficient. An RCS cooldown must be initated first to establish at least 30°F subcooling.

D. Plausible because any RCP *may* be started and because an RCP should be started to ensure that PZR Spray is available for RCS pressure control and to mix reactor coolant to uniform temperatures.

Incorrect because spray valves will not be effective in pressure control with the PZR water-solid. and because subcooling is insufficient.

K/A - 008 G2.4.6:

(Pressurizer Vapor Space Accident) Knowledge symptom based EOP mitigation strategies.

Tier: 1
Group: 1

Importance Rating: SRO 4.7

Technical References:

- ES-1.2 WOG Background Document for Step 12 (HES12BG.doc, HP-Rev 2, 4/30/05, p44, 94-95)
- SOP-101, Precaution 2.a.6.d, Rev 26
- AB-4 (p39, Rev 12)
- EOP-2.1, Step 9 (Rev 13)

Proposed references to be provided to applicants during examination:

None

Learning Objective: EOP-2.1-05 & 07

Question History: NEW

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

The KA is matched because the operator must demonstrate knowledge of the EOP mitigation strategies during a PZR Vapor Space Accident (i.e. it is acceptable to permit RCP restart even without a steam bubble in the PZR, and the reason for starting an RCP).

SRO Only Justification:

The question is SRO-Only because the operator must assess plant conditions, and know the content of mitigation procedures, including the basis for this content.

5. 013 G2.2.42 001

Given the following plant conditions:

- 100% power
- RM-A4, Reactor Building Purge Exhaust Monitor, is Out-of-Service.

Which ONE (1) of the following identifies all of the Technical Specification and/or ODCM Specification of Controls which will prevent placing the Alternate Purge Exhaust System in operation?

REFERENCES PROVIDED

A. LCO 3.3.2, ESFAS Instrumentation; AND

LCO 3.3.3.1, Radiation Monitoring Instrumentation.

B.✓ LCO 3.3.2, ESFAS Instrumentation; AND

ODCM 1.2.1, Radioactive Gaseous Effluent Monitoring Instrumentation.

C. LCO 3.3.3.1, Radiation Monitoring Instrumentation; AND

ODCM 1.2.1, Radioactive Gaseous Effluent Monitoring Instrumentation.

D. ODCM 1.2.1, Radioactive Gaseous Effluent Monitoring Instrumentation ONLY.

A. Plausible because the 1st part is correct. Also plausible because LCO 3.3.3.1 is related to radiation monitoring.

Incorrect. According to T.S. LCO 3.3.3.1, the Radiation Monitoring Instrumentation channels shown in Table 3.3-6 shall be OPERABLE. Table 3.3-6, Instrument 2.b.i, Containment – Gaseous Activity – Purge & Exhaust Isolation (RM-A4), a minimum of one channel must be OPERABLE in Mode 6. Therefore, the LCO does NOT apply at this time.

B. CORRECT. According to T.S. LCO 3.3.2, the ESFAS Instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE. Table 3.3-3, Functional Unit 3C2, Purge and Exhaust Isolation, a minimum of two channels required to trip must be OPERABLE in Modes 1-4, whenever purge exhaust is OPEN. According to GS-9, Table GS9.4, these two channels (i.e. monitors) are RM-A2, RB Sample Line, and RM-A4, RB Purge Exhaust Monitor, both of which will automatically close the Alternate Purge Supply and Exhaust System Isolation Valves on high radiation. If RM-A4 is inoperable, According to TS Table 3.3-3, ACTION 17 must be addressed. This ACTION statement requires that with less than the minimum channels OPERABLE, operation may continue provided the containment supply and exhaust valves are maintained closed. According to ODCM Requirement 1.2.1, the Radioactive Gaseous Effluent Monitoring Instrumentation channels shown in Table 1.2-1 shall be OPERABLE. Table 1.2-1, Functional Unit 3, Reactor Building Purge System, a minimum of one channel shall be OPERABLE at all times during releases from this pathway. According to Table 1.2-1, this channel is RM-A4, RB Purge Exhaust Monitor, which has separate requirements for the gas sampler including alarm and automatic termination of the release, the iodine sampler, the particulate sampler, the flow measurement devices, and the sampler flow rate measuring device . If the gas sampler (RM-A4) is inoperable, ACTION 10 must be addressed. This ACTION statement requires that, with less than the minimum channels OPERABLE, immediately suspend purging of radioactive effluents via this pathway.

C. Plausible because the 2nd half is correct. Also plausible because LCO 3.3.3.1 is related to radiation monitoring.

Incorrect. According to TS LCO 3.3.3.1, the Radiation Monitoring Instrumentation channels shown in Table 3.3-6 shall be OPERABLE. Table 3.3-6, Instrument 2.b.i, Containment – Gaseous Activity – Purge & Exhaust Isolation (RM-A4), a minimum of one channel must be OPERABLE in Mode 6. Therefore, the LCO does NOT apply at this time.

D. Plausible because ODCM 1.2.1 is part of the correct answer.

Incorrect since <u>both</u> LCO 3.3.2 and ODCM 1.2.1 prohibit placing the Alternate Purge Supply and Exhaust System in operation.

K/A - 013 G2.2.42:

(Engineered Safety Features Actuation) Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Tier: 2
Group: 1

Importance Rating: SRO 4.6

Technical References:

- TS LCO 3.3.2, Table 3.3-3, Action 17
- GS-9, Table GS9.4
- ODCM Requirement 1.2.1, Table 1.2-1, Action 10
- TS LCO 3.3.3.1, Table 3.3-6

Proposed references to be provided to applicants during examination:

- T.S. 3.3.2
- T.S. 3.3.3.1
- ODCM 1.2.1

Learning Objective: SB-4-19

Question History: NEW

10 CFR Part 55 Content: 43(b)(2)

K/A Match:

The KA is matched because the operator must demonstrate the ability to recognize system parameters (RM-A4 inoperable) that are entry-level conditions for Technical Specifications associated with the ESFAS.

SRO Only Justification:

The question is SRO-Only because not only does the operator need to know the requirements of the LCO, but the operator needs to know action associated with the inoperablity of RM-A4, specifically that with RM-A4 inoperable, radioactive releases via this pathway can NOT continue.

6. 015 AA2.10 001

0/10Given the following plant conditions:

- 8% power
- CCW flow to the RCP Bearing Oil Coolers is lost.
- The crew is addressing the appropriate Annunciator Response Procedures.
- During the recovery, the following RCP parameters are observed EIGHT (8) minutes after the loss of flow:

	A	B	C
• Highest Motor Bearing Temperature	189°F	194°F	196°F
• Lower Seal Water Bearing Temperature	105°F	105°F	103°F
• Seal Water Outlet Temperature	100°F	104°F	102°F

Which ONE (1) of the following identifies the RCP(s) that have met their trip criteria AND the specific procedure(s) that are REQUIRED to be addressed in accordance with SOP-101, *Reactor Coolant System*?

A.✓ ONLY RCP C;

GOP-4B, *Power Operation (Mode 1 Descending);* and GOP-5, *Reactor Shutdown from Startup to Hot Standby (Mode 2 to Mode 3).*

B. ONLY RCPs B and C;

GOP-4B, *Power Operation (Mode 1 Descending);* and GOP-5, *Reactor Shutdown from Startup to Hot Standby (Mode 2 to Mode 3)*

C. ONLY RCP C;

EOP-1.0, *Reactor Trip/Safety Injection Actuation*

D. ONLY RCPs B and C;

EOP-1.0, *Reactor Trip/Safety Injection Actuation*

A. CORRECT. According to ARP-001, Rev 5, XCP-602, 2-3, Supplemental Action 1, if Motor Bearing Temperature reaches, affected RCPs must be stopped. If the RCPs must be stopped while in Mode 4 or above, and if power is > than 10%, trip the Reactor and secure the RCPs per SOP-101. If power is < 10%, the operator is directed to secure the RCPs per SOP-101. When this is done, T.S. LCO 3.4.1 is NOT satisfied. The LCO requires that when in Modes 1 and 2, all RCS loops be in operation. The ACTION states that with < all RCS Loops in operation, be in at least Hot Standby within 1 hour. In order to place the plant in Hot Standby, GOPs 4B & 5 must be used.

B. Plausible because the second part is correct. Plausible because the 1st part *would* be correct if RCP 'B' temp limits had been exceeded. According to ARP-001, Rev 5, XCP-602, 2-3, Supplemental Action 1, if flow is lost to the bearing coolers, the RCPs must be stopped before motor bearing temperature reaches 195°F.

Incorrect because <u>only</u> RCP 'C' has exceeded the 195°F limit on Motor Bearing Temperature.

C. Plausible because the 1st part is correct. 2nd part is also plausible since EOP-1.0 would be used if power were greater than 38% (for 1 RCP) <u>and</u> could possibly be *conservatively* used if 10 minutes had been exceeded and all 3 RCPs required tripping. Additionally, initiating a Reactor Trip *would* place the unit in Hot Standby, which is the desired end state.

2nd part is incorrect because the ARP specifically addresses the use of a Reactor Trip ONLY when power is > 10%.

D. Plausible because the 1st part *would* be correct if RCP 'B' temp limits had been exceeded. According to ARP-001, Rev 5, XCP-602, 2-3, Supplemental Action 1, if flow is lost to the bearing coolers, the RCPs must be stopped before motor bearing temperature reaches 195°F. 2nd part is also plausible since EOP-1.0 would be used if power were greater than 38% (for 1 RCP) <u>and</u> could possibly be *conservatively* used if 10 minutes had been exceeded and all 3 RCPs required tripping. Additionally, initiating a Reactor Trip *would* place the unit in Hot Standby, which is the desired end state.

Incorrect because <u>only</u> RCP 'C' has exceeded the 195°F limit on Motor Bearing Temperature. 2nd part is incorrect because the ARP specifically addresses the use of a Reactor Trip ONLY when power is > 10%.

K/A - 015 AA2.10:

Ability to (a) predict the impacts of the following on the RCP Malfunctions and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: When to secure RCPs on loss of cooling or seal injection.

Tier: 1
Group: 1

Importance Rating: SRO 3.7

Technical References:

- ARP-001, Rev 5, XCP-602 AP2-3, Supplemental Action 1 and 6
- TS LCO 3.4.1 and its Basis

Proposed references to be provided to applicants during examination:

None

Learning Objective: AOP-118.1-09

Question History:

MODIFIED (Although written "from scratch", this question is similar enough to Closed Reference questions AOPS 379, 360, 359, 312, 71 to be classified as MODIFIED)

10 CFR Part 55 Content: 43(b)(2),(5)

K/A Match:

The KA is matched because the operator must demonstrate the ability to determine when an RCP must be tripped, by interpreting a set of given conditions, during a loss of cooling to the RCPs.

SRO Only Justification:

The question is SRO-Only because it requires the operator to recall a strategy or action that is written in a plant procedure (action to take when all RCPs must be tripped), and when to take this strategy (< 10% power), AND the question involves application of required TS ACTION (TS LCO 3.4.1).

7. 026 G2.2.44 001

Given the following plant conditions:

- Mode 1
- B2 Maintenance Week is in progress
- Annunciator CCW SRG TK LVL HI/LO/LO-LO (XCP-601, 1-1) has just actuated.
- The NROATC has determined that it is a LOW level.
- CCW Surge Tank 'A' level is decreasing slowly.

Given the following equipment nomenclature:

- LCV-7088, MU TO SRG TK (B SIDE)
- LCV-7090, MU TO SRG TK (A SIDE)
- PVG-9627A, SW TO CC LOOP A

Based on the existing conditions, which ONE (1) of the following describes the response of the CCW System makeup valves and the actions that must be taken in accordance with XCP-601, 1-1?

A. ONLY LCV-7088 and LCV-7090 are open;

Swap the running charging pump per SOP-102 and apply actions for T.S. 3.1.2.2, Flow Paths - Operating

B.✓ ONLY LCV-7088 and LCV-7090 are open;

Swap CCW active loops per SOP-118 and apply actions for T.S. 3.7.3, Component Cooling Water System

C. LCV-7088, LCV-7090, and PVG-9627A are all open;

Swap the running charging pump per SOP-102 and apply actions for T.S. 3.1.2.2, Flow Paths - Operating

D. LCV-7088, LCV-7090, and PVG-9627A are all open;

Swap CCW active loops per SOP-118 and apply actions for T.S. 3.7.3, Component Cooling Water System

A. Plausible because the 1st part is correct. According to XCP-601, 1-1, Automatic Action 1.b., LCV-7088 and LCV-7090 open on a LOW level. 2nd part is plausible because it is partially correct - Supplemental Action 6 of XCP-601, 1-1, requires the inservice Charging Pump to be transferred to the active CCW loop per SOP-102.

The 2nd part is incorrect because the Action of T.S. 3.1.2.2 does not apply. While the original inservice Charging Pump will not be available as a Boration Flow Path, the LCO requires only two of the three identified flowpaths to be OPERABLE. Since the gravity drain path and the other charging pump are available, the LCO is met, and no actions apply.

B. CORRECT. According to XCP-601, 1-1, Automatic Action 1.b., LCV-7088 and LCV-7090 open on a LOW level. Supplemental Action 5 of this ARP states: "On low level or low-low level, if level continues to decrease, establish Train B as the active loop per SOP-118. With one train inoperable, refer to Tech Spec 3.7.3.

C. Plausible because the 1st part of the 1st part is correct. According to XCP-601, 1-1, Automatic Action 1.b., LCV-7088 and LCV-7090 open on a LOW level. Also plausible because PVG-9627A will eventually open below the low-low level if the leak is not isolated. Incorrect since PVG-9627A will not open on a LOW level. 2nd part is plausible because it is partially correct - Supplemental Action 6 of XCP-601, 1-1, requires the inservice Charging Pump to be transferred to the active CCW loop per SOP-102.

The 2nd part is incorrect because the Action of T.S. 3.1.2.2 does not apply. While the original inservice Charging Pump will not be available as a Boration Flow Path, the LCO requires only two of the three identified flowpaths to be OPERABLE. Since the gravity drain path and the other charging pump are available, the LCO is met, and no actions apply.

D. Plausible because the 1st part of the 1st part is correct. According to XCP-601, 1-1, Automatic Action 1.b., LCV-7088 and LCV-7090 open on a LOW level. Also plausible because PVG-9627A will eventually open below the low-low level if the leak is not isolated.

Incorrect since PVG-9627A will not open on a LOW level. Supplemental Action 5 of this ARP states: "On low level or low-low level, if level continues to decrease, establish Train B as the active loop per SOP-118. With one train inoperable, refer to Tech Spec 3.7.3.

K/A - 026 G2.2.44:

(Loss of Component Cooling Water) Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Tier: 1
Group: 1

Importance Rating: SRO 4.4

Technical References:

- XCP-601, 1-1 (Rev 5)
- T.S. 3.7.3
- T.S. 3.1.2.2

Proposed references to be provided to applicants during examination:

None

Learning Objective: IB-2-21

Question History:

MODIFIED (Although written "from scratch", this question is similar enough to Closed Reference question CCW SYSTEM 136 to be classified as MODIFIED)

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

The KA is matched it involves interpretation of CR indications to verify status of the CCW system (CCW Surge Tank alarm)and understanding of how actions affect the plant (application of ARP Supplental Actions and INOPERABLE CCW Train)

SRO Only Justification:

The question is SRO-Only because the operator must assess the stated conditions, which are abnormal, and apply a detailed step from the ARP Supplemental Actions (*specific* mitigation strategies).

8. 034 A2.03 008

Which ONE (1) of the choices below completes the following statement?

To ensure that a fuel assembly has not been mispositioned during core reload, REP-107.013, *Core Reload*, requires the Refueling SROs to check that the fuel assembly reference hole in the top nozzle is oriented toward the __________, and the Refueling SROs are required to sign the permanent copy of the Material Transfer Form ___________.

A. Northeast

after each fuel assembly movement

B.✓ Northeast

prior to the end of their shift

C. East-Southeast

after each fuel assembly movement

D. East-Southeast

prior to the end of their shift

A. Plausible because the 1st part is true (per Section 7.3.5 of REP-107.013). 2nd part is plausible because the SRO will sign off the RB copy at the end of each movement.

Incorrect because the Refueling SRO signs the Control Room (permananent) Material Transfer Form, at the end of shift, NOT after each movement.

B. CORRECT. IAW Section 7.3.5 of REP-107.013, the hole orientation is towards the NE. IAW Section 3.2.1, the Refueling SRO will sign the Material Transfer Form.

C. 2nd part is plausible because the SRO will sign off the RB copy at the end of each movement.. East-Southeast is plausible because this is a direction addressed in HPP-709 regarding avoiding gaseous releases when the wind is from this direction.

Incorrect because the Refueling SRO signs the Control Room (permananent) Material Transfer Form, at the end of shift, NOT after each movement. Also incorrect because the assembly orientation is with the hole towards the NE, NOT E-SE.

D. Plausible because the second part is correct. East-Southeast is plausible because this is a direction addressed in HPP-709 regarding avoiding gaseous releases when the wind is from this direction.

Incorrect because the assembly orientation is with the hole towards the NE, NOT E-SE.

K/A - 034 A2.03:

Ability to (a) predict the impacts of the following on the Fuel Handling Equipment and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Mispositioned fuel element.

Tier: 2
Group: 2

Importance Rating: SRO 4.0

Technical Reference:

- REP-100.001, Attachment I, Material Transfer Form
- REP-107.013, Section 3.2.1, Section 7.3.5

Proposed references to be provided to applicants during examination:

- None

Learning Objective: REFUELING SRO-12, 14

Question History: NEW

10 CFR Part 55 Content: 43(b)(6)

K/A Match:

Matches the K/A in that it tests ability to determine that an assembly has been mispositioned (via verifying/signing the Material Transfer Form) and the correct orientation of a properly positioned fuel element.

SRO Only Justification:

SRO Only in that it tests knowledge of Refueling SRO-specific responsibilities.

9. 037 G2.4.8 001

Given the following plant conditions:

- 100% power
- The crew has implemented AOP-112.2, *Steam Generator Tube Leak NOT Requiring SI.*
- A plant shutdown has been initiated.
- Due to the S/G tube leak, a Reactor Trip is initiated.
- The crew has transitioned to EOP-1.1, *Reactor Trip Recovery*, checked Feedwater status, and checked RCS Tavg trending to 557°F (Step 3).

Which ONE (1) of the following describes the procedure flowpath?

A. Remain in EOP-1.1, complete the EOP, then GO TO AOP-112.2.

B. GO TO AOP-112.2, complete the AOP, then GO TO EOP-1.1, Step 4.

C. Remain in EOP-1.1 and REFER TO AOP-112.2 and perform actions of AOP-112.2 as time allows.

D.✓ GO TO AOP-112.2 and REFER TO EOP-1.1 and perform actions of EOP-1.1 as time allows.

A. Plausible because it is consistent with general hierarchy of EOP-AOP.

Incorrect because EOP-1.1 would not be completed before transitioning to AOP-112.2.

B. Plausible because the 1st part is true.

Incorrect because the crew is supposed to implement non-conflicting steps of EOP-1.1 as time allows, while implementing AOP-112.2.

C. This is plausible because the 1st part would be true if the crew had just entered EOP-1.1.

Incorrect because the crew should GO TO, not REFER TO, AOP-112.2.

D. CORRECT. According to OAP-103.4, EOP/AOP User's Guide, Step 6.4.e, Rev 0, the use of the phrases "Go To" and "Refer To" are used to determine which procedure constitutes the primary flowpath to target the correct recovery action. The phrase "Go To" is used to leave the procedure or step in progress and transition to the required procedure or step for continued recovery actions, with the procedure transitioned to becoming the new primary procedure in effect. The phrase "Refer To" is used to continue with the procedure in progress using the referenced procedure as a guideline to accomplish a specific action. In this case, the referenced procedure is used concurrently with the primary procedure in progress. Upon exiting EOP-1.0, Reactor Trip/Safety Injection Actuation, the operator is directed to Go To EOP-1.1, Step 1, at the Step 5 Alternative Action (Rev 22), making EOP-1.1 the primary procedure in progress. The first three steps of EOP-1.1 (Rev 15) direct the operator to (1) announce plant conditions over the page system, (2) check FW status, and (3) check RCS Temperature. Then the operator is provided with a Note prior to step 4 and the step itself. The Note states that if a transition is made to AOP-112.2, the steps of EOP-1.1 which do NOT conflict with AOP-112.2 should be completed as time allows. Then step 4 directs the operator to return to AOP-112.2, step 7, if EOP-1.0 was entered from AOP-112.2 (which it was), making AOP-112.2 the primary procedure in progress.

K/A - 037 G2.4.8:

(Steam Generator Tube Leak) Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Tier: 1
Group: 2

Importance Rating: SRO 4.5

Technical References:

- OAP-103.4, EOP/AOP User's Guide, Step 6.4.e, Rev 0
- EOP-1.0, Step 5, Rev 22
- EOP-1.1, Step 1-4, Rev 15
- AOP-112.02, Rev 4

Proposed references to be provided to applicants during examination:

None

Learning Objective: AOP-112.2-06; EOP-1.1-05

Question History: NEW

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

The KA is matched because the operator must demonstrate knowledge of how abnormal operating procedures (AOP-112.2) are used in conjunction with EOPs (EOP-1.1) with the identified rules of usage.

SRO Only Justification:

The question is SRO-Only because the operator must recall that the strategy for mitigating a Steam Generator Tube Leak makes AOP-112.02 the primary procedure in progress after the first three steps of EOP-1.1 are performed, and renders the WOG based EOP to a secondary or referenced procedure status.

10. 054 AA2.04 002

Given the following plant conditions:

- 100% power
- A loss of ALL Main Feedwater occurs.
- A manual Reactor Trip was initiated and the crew is implementing EOP-1.1, *Reactor Trip Recovery.*
- MDEFP 'A' and 'B' fail during the recovery.
- Instrument Air Compressor (IAC) 'A' trips and IAC 'B' cannot be started.
- The Supplemental Air Compressor is aligned for normal operations.
- NR Level in all SGs is 67% and rising.
- The CRS directs SG levels to be controlled between 60% and 65%.

Given the *current* conditions, which ONE (1) of the following identifies how the level in all SGs will be controlled AND the proper procedure to be used?

A. MCB throttling of TDEFP FCVs;

EOP-15.2, *Response to Steam Generator High Level.*

B.✓ MCB throttling of TDEFP FCVs;

EOP-1.1, *Reactor Trip Recovery.*

C. Local throttling of TDEFP FCVs;

EOP-15.2, *Response to Steam Generator High Level.*

D. Local throttling of TDEFP FCVs;

EOP-1.1, *Reactor Trip Recovery.*

A. Plausible because the 1st part is correct - with no indications that a loss of I.A. has occurred, the operators should be able to control FCVs from the MCB. Also plausible because 67% is close to the EOP-15.2 criteria of 87%.

Incorrect because the entry condition for EOP-15.2 is 87% Narrow Range Level (see EOP-12.0, Rev 12, Attachment 3), which does NOT exist at the time of the given conditions in the stem. Also incorrect because the operator will apply the Continuous Action to maintain SG level between 40-60% at step 9 of EOP-1.1, well before control would be established in EOP-15.2.

B. CORRECT. With no indications that a loss of I.A. has occurred, the operators should be able to control FCVs from the MCB. At Step 9 of EOP-1.1 (Rev 15), the operator is directed to verify SG levels between 40-60%. This is a Continuous Action step (i.e. marked with an asterisk), and therefore when the C SG level rises to > 60%, Step 9b will be required. This step directs the operator to control EFW flow to maintain SG Level between 40-60%.

C. Plausible because local operator action *is* an option if I.A. is lost or MCB actions are ineffective. Given the failure of the IACs in bullet #5, local action is plausible. Also plausible because 67% is close to the EOP-15.2 criteria of 87%.

2nd part is incorrect because the entry condition for EOP-15.2 is 87% Narrow Range Level (see EOP-12.0, Rev 12, Attachment 3), which does NOT exist at the time of the given conditions in the stem. Also incorrect because the operator will apply the Continuous Action to maintain SG level between 40-60% at step 9 of EOP-1.1, well before control would be established in EOP-15.2. Also incorrect because the 1st part is wrong - MCB throttling is available because both the Supplemental IAC and the Diesel IAC are available.

D. Plausible because the 2nd part is correct - At Step 9 of EOP-1.1 (Rev 15), the operator is directed to verify SG levels between 40-60%. Also plausible because local operator action *is* an option if I.A. is lost or MCB actions are ineffective. Given the failure of the IACs in bullet #5, local action is plausible.

Incorrect because the 1st part is wrong - MCB throttling is available because both the Supplemental IAC and the Diesel IAC are available.

K/A - 054 AA2.04:

Ability to determine and intepret the following as they apply to a Loss of Main Feedwater (MFW): Proper operation of AFW pumps and regulating valves.

Tier: 1
Group: 1

Importance Rating: SRO 4.3

Technical References:

- LCO 3.7.1.2, Action Statement a/b
- EOP-12.0, Rev 12, Attachment 3
- OAP-103.4, EOP/AOP User's Guide, Step 6.14.d, Rev 0
- IB-3 (p32-33, Rev 17)
- Step 9 of EOP-1.1 (Rev 15)

Proposed references to be provided to applicants during examination:

None

Learning Objective: EOP-1.1-06

Question History: NEW

10 CFR Part 55 Content: 43(b)(2), (5)

K/A Match:

The KA is matched because the operator must demonstrate the ability to determine the proper operation of EFW FVCs and determine actions (procedural actions for valve control) to take for both.

SRO Only Justification:

The question is an SRO-Only question because it requires the operator to assess plant conditions and select a recovery procedure (including evaluation of Yellow Path applicability).

11. 062 G2.2.36 003

Given the following plant conditions:

- 100% power
- During a routine surveillance, it is revealed that Train 'B' of SSPS will NOT automatically actuate RB Spray.
- Train 'B' of ESFAS Instrumentation, Functional Unit (Item) 2.b, Reactor Building Spray – Automatic Actuation Logic and Associated Relays, is declared INOPERABLE.
- LCO 3.3.2, ACTION 14, is entered, placing the plant in a shutdown LCO.

Which ONE (1) of the following planned activities must be rescheduled until the operability of SSPS Train 'B' has been restored?

A. Swap Charging Pump 'C' to Charging Pump 'B'.

B.✓ Swap Service Water Pump 'A' to Service Water Pump 'C'.

C. Perform the quarterly surveillance on Residual Heat Removal Pump 'B'.

D. Replace the breaker for MVG-8801B, HI HEAD TO COLD LEG INJ.

A. Plausible because swapping pumps is a routine maintenance activity under normal conditions.

Incorrect because it does NOT affect the OPERABILITY of Train 'A' and would be allowed.

B. CORRECT. According to the T.S. LCO 3.3.2 Basis, entry into ACTION statements 12, 14, 21 or 25 is NOT a typical, pre-planned evolution during power operation, other than for surveillance testing; and generally are entered due to an equipment failure. If these ACTION statements are entered some restrictions will apply. For instance, to preserve ATWS mitigation capability, activities that degrade the availability of the EFW System, RCS Pressure Relief System, AMSAC or Turbine Trip should NOT be scheduled when a logic train is inoperable for maintenance. Additionally, to preserve the LOCA mitigation capability, one complete ECCS train that can be actuated automatically must be maintained when a logic train is inoperable for maintenance. Furthermore, to preserve reactor trip and safeguards actuation capability, activities that cause master relays or slave relays in the available train to be unavailable and activities that cause analog channels to be unavailable should not be scheduled when a logic train is inoperable for maintenance. Finally, any activities on electrical systems (e.g. AC and DC power) and cooling systems (e.g. service water and CCW) that support the systems or functions listed in the first three conditions should NOT be scheduled when a logic train is inoperable for maintenance. That is, one complete train of a function that supports a complete train of a function noted above must be available. With this in mind, the cycling of the PORV Block Valve can NOT be performed because it is an activity that degrades the availability of the RCS Pressure Relief System. Secondly, the swap of the A Train Service Water Pump to the C Service Water Pump can NOT be performed because it is an activity on electrical systems (e.g. AC power) and cooling systems (e.g. service water) that supports the systems or functions listed in the first three conditions, such EFW, and ECCS. On the other hand, the swap of the B Train Charging Pump to the C Charging Pump may be performed because it does NOT affect the A Train of ECCS, which can still be automatically actuated. Additionally, the monthly Surveillance on the B Diesel Generator can be performed because one complete Train (e.g. A Train) that supports a complete train of the function noted above is still available.

C. Plausible because it is a routine evolution that *could* be performed under normal conditions.

Incorrect because the quarterly surveillance on RHR Pump 'B' *can* be performed because one complete Train (e.g. A Train) that supports a complete train of the function noted above is still available.

D. Plausible because it is an activity that *could* be performed under normal conditions.

Incorrect because the breaker replacement *can* be performed because one complete Train (e.g. A Train) that supports a complete train of the function noted above is still available.

K/A - 062 G2.2.36:

(Loss of Nuclear Service Water) Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions of operations.

Tier: 1
Group: 1

Importance Rating: SRO 4.2

Technical Reference:

- TS 3.3.2 & Basis.

Proposed references to be provided to applicants during examination:

None

Learning Objective: SB-4-19, 21

Question History: NEW

10 CFR Part 55 Content: 43(b)(2)

K/A Match:

The KA is matched because the operator must apply detailed knowledge of a T.S. bases (impact of SW swapover) to analyze the effect of maintenance activities, such as degraded power sources, and others, on the status of limiting conditions for operations (LCO 3.3.2).

SRO Only Justification:

The question is SRO-Only because it requires that the operator have knowledge of the basis of TS LCO 3.3.2 (not a Safety Limit) necessary to analyze the TS action required.

12. 062 G2.4.6 001

Given the following plant conditions:

- 100% power
- A sustained total loss of offsite power has occurred.
- Both ESF Diesel Generators have failed to start automatically and manually.
- The crew has implemented EOP-6.0, *Loss of All ESF AC Power.*
- ESF equipment has been placed in PULL-TO-LOCK.
- Continued attempts to start the DGs have been unsuccessful.
- Subsequently, at Step 11 of EOP-6.0, the crew successfully energizes Bus 1DA via XTF-5052, ALTERNATE AC POWER SUPPLY TRANSFORMER.

Which ONE (1) of the following identifies the breaker that, when closed, will restore power to bus 1DA, AND identifies the action required in accordance with EOP-6.0, when the Bus is restored?

A.✓ The 1DA NORM Feed Breaker;

GO TO Step 30 and stabilize SG pressure.

B. The 1DA ALT Feed Breaker;

GO TO Step 30 and stabilize SG pressure.

C. The 1DA NORM Feed Breaker;

GO TO EOP-1.0, *Reactor Trip/Safety Injection Actuation*, Step 1.

D. The 1DA ALT Feed Breaker;

GO TO EOP-1.0, *Reactor Trip/Safety Injection Actuation*, Step 1.

A. CORRECT. The crew is operating at Step 11 Alternate Action of EOP-6.0 (Rev 21), which directs the operator to refer to SOP-304, Section V.A, (Rev 11) to supply offsite power from the Alternate AC Power Supply. SOP-304 provides a Caution prior to step 2.8 indicating that only one AC ESF Bus can be supplied using that Alternate AC Power Supply at any one time. A series of steps is provided in SOP-304, Step 2.8.a.4, to energize the 1DA ESF Bus. This section includes a step to close the 1DA NORM Feeder Breaker which will re-energize the bus. According to Step 2.8.b.4, in order to re-energize the 1DB ESF Bus the 1DB ALT Feeder Breaker must be closed. According to Caution - Step 8 just prior to Step 8 of EOP-6.0, when power is restored to either ESF Bus, recovery should continue with Step 30 to minimize the deterioration of plant conditions.

B. Plausible because the 2nd part is correct. Also plausible because the 1DB ALT Feeder Breaker must be closed to re-energize the 1DB ESF Bus.

Incorrect because the 1st part is wrong – SOP-304 provides a Caution prior to step 2.8 indicating that only one AC ESF Bus can be supplied using that Alternate AC Power Supply at any one time. A series of steps is provided in SOP-304, Step 2.8.a.4, to energize the 1DA ESF Bus. This section includes a step to close the 1DA NORM Feeder Breaker which will re-energize the bus. .

C. Plausible because the 1st part is correct - a series of steps is provided in SOP-304, Step 2.8.a.4, to energize the 1DA ESF Bus. This section includes a step to close the 1DA NORM Feeder Breaker which will re-energize the bus. 2nd part is plausible because it is the <u>next</u> step in the procedure if the Caution is misapplied.

Incorrect because, per CAUTION - Step 8, the crew is directed to Step 30, which stabilizes SG pressures, NOT to Step 12, establish RB I.A.

D. 1st part is plausible because the ALT feeder breaker would be used to restore Bus *1DB* from the Alternate AC source. 2nd part is plausible because it is the <u>next</u> step in the procedure if the Caution is misapplied.

Incorrect because the first part is wrong - a series of steps is provided in SOP-304, Step 2.8.a.4, to energize the 1DA ESF Bus. This section includes a step to close the 1DA NORM Feeder Breaker which will re-energize the bus. Also incorrect because, per CAUTION - Step 8, the crew is directed to Step 30, which stabilizes SG pressures, NOT to Step 12, establish RB I.A.

K/A - 062 G2.4.6:

(AC Electrical Distribution) Knowledge of symptom based EOP mitigation strategies.

Tier: 2
Group: 1

Importance Rating: SRO 4.7

Technical References:

- EOP-6.0 Caution 6, Step 8
- SOP-304, Section V.A, Step 2.8, Rev 11
- TS Basis for LCO 3.8.1

Proposed references to be provided to applicants during examination:

None

Learning Objective: EOP-6.0-05

Question History: NEW

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

The KA is matched because the operator must demonstrate have system based knowledge of the AC Distribution System (Breaker nomenclature), and knowledge of the EOP mitigation strategies (specific procedure flowpath).

SRO Only Justification:

The question is SRO-Only because the operator must recall that the specific strategy (and the specific action taken) for moving forward to step 30 is written into EOP-6.0, when past Step 10, and an ESF Bus is restored with the Alternate AC Power Supply.

13. 064 A2.13 001

Given the following plant conditions:

- 100% power
- A routine surveillance of Diesel Generator 'A' is in progress.
- Diesel Generator 'A' is operating in parallel with offsite power.
- Service Water Pump 'C' is aligned to Train 'A' but not running.
- Subsequently, the feeder breaker for Bus 1EA trips OPEN.
- The following DG 'A' conditions are observed:
    - The DG 'A' lube oil temperature is 160°F and rising at 1.0°F/minute.
    - The DG 'A' coolant temperature is 178°F and rising at 1.5°F/minute.

Which ONE (1) of the following identifies the approximate time frame, within which, Diesel Generator 'A' will trip if NO action is taken, AND the actions that are required be taken within AOP-117.1, *Loss of Service Water*?

A. 4-8 minutes;

Transfer Service Water System loads to Train 'B' in accordance with ARP-001, XCP-604.

B. 4-8 minutes;

Start SWP 'C' and open PVG-3105A, FS TO DG A, in accordance with SOP-117.

C✓ 10-12 minutes;

Transfer Service Water System loads to Train 'B' in accordance with ARP-001, XCP-604.

D. 10-12 minutes;

Start SWP 'C' and open PVG-3105A, FS TO DG A, in accordance with SOP-117.

A. Plausible because the 2nd part is correct. Also plausible because the Lube Oil High Temp Alarm will come in within 7 minutes [167-160=7/1=7] and the High Jacket Water Temp Alarm will come in in 4.7 minutes [185-178=7/1.5=4.67].

Incorrect because the 1st part is wrong – The heat rate on the DG will NOT result in a trip within 7 minutes – See trip calculations in Choice C.

B. Plausible because the Lube Oil High Temp Alarm will come in within 7 minutes [167-160=7/1=7] and the High Jacket Water Temp Alarm will come in in 4.7 minutes [185-178=7/1.5=4.67]. Also plausible because ARP-604, 3-4 Supplemenatl Action 1 restores cooling to the DG via PVG-3105 and because Corrective Action 2 of the same ARP directs the crew to place the spare SWP in service per SOP-117.

Incorrect because the 1st part is wrong – The heat rate on the DG will NOT result in a trip within 7 minutes – See trip calculations in Choice C. Also incorrect because SWP 'C' cannot be started under the given conditions in the stem.

C. CORRECT. According to GS-2, Table GS2.2 (p81, Rev 15), the A Service Water Pump is powered from 7.2 KV Stub Bus 1EA and therefore, will be de-energized when the feeder breaker for ESF Stub Bus 1EA trips OPEN. According to IB-5 (p49, Rev 21) (and ARP XCX-5201, 1-2) , the Diesel will trip on high lube oil temperature of 175°F and high jacket coolant temperature of 195°F, when the DG is operated in the TEST Mode (Which it is when conducting a routine surveillance). At the rate indicated, the High Lube Oil Temperature will be reached in 15 minutes [(175°F -160°F) x minute/1°F = 15 minutes], and the High Jacket Coolant Trip will be reached in 11.3 minutes [(195°F -178°F) x 2 minutes/3°F = 11.3 minutes]. According to AOP-117.1 (Rev 3), the entry conditions for the AOP are met. The operator will refer to the appropriate ARPs to restore Service Water, one of which will be ARP-001, XCP-604, AP3-4, DG A CLR SW FLO LO TEMP HI. This ARP has a Supplementary Action (1) that will restore SW to the operating DG by opening PVG-3105A to provide water from the Fire System. This will end the immediate threat to the loaded DG which is operating without cooling water. Additionally, Steps 5 and 6 of AOP-117.1 will direct the operator to transfer Service Water System loads to Train B.

D. Plausible because the 1st part is correct - see trip calculations in Choice C. Also plausible because ARP-604, 3-4 Supplemenatl Action 1 restores cooling to the DG via PVG-3105 and because Corrective Action 2 of the same ARP directs the crew to place the spare SWP in service per SOP-117.

Incorrect because SWP 'C' cannot be started under the given conditions in the stem.

K/A - 064 A2.13:

Ability to (a) predict the impacts of the following on the Emergency Diesel Generator and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Consequences of opening auxiliary feeder bus (ED/G sub supply).

Tier: 2
Group: 1

Importance Rating: SRO 2.8

Technical Reference:

- GS-2, Table GS2.2 (p81, Rev 15)
- IB-5 (p49, Rev 21)
- AOP-117.1 (Rev 3)

Proposed references to be provided to applicants during examination:

None

Learning Objective: IB-5-21; AOP-117.1-06

Question History: NEW

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

The KA is matched because the operator must demonstrate the ability to predict the impact opening auxiliary feeder bus (ED/G sub-supply or ESF Stub Bus 1EA) on the ED/G system (Requires a DG trip if no action taken); and (b) based on those predictions, identify the steps taken within AOP-117.1 (& ARPs) to mitigate and control the event.

SRO Only Justification:

The question is SRO-Only because the operator must recall detailed strategy in ARP 604 & AOP-117.1 for dealing with a loss of SW (i.e. Use ARPs and shift SW loads to the operating train).

14. 068 A2.04 002

Given the following plant conditions:

- A Waste Monitor Tank #2 release is in progress.
- The following annunciators have actuated:
  - LIQ WST DISCH RM-L9 HI RAD (XCP-644, 2-5)
  - LIQ WST DISCH RM-L9 TRBL (XCP-644, 2-6)
- Release flow rate remains at 55 gpm.

Which ONE (1) of the following completes the statement below?

The operator must close _________as required by ARP XCP-644. Failure to take this action may result in the dose commitment to an individual (general public) at the ___________ from radioactive materials in liquid effluents to exceed the ODCM limit to the total body during any calendar quarter.

A. RCV-018, Liquid Waste Control Valve;

Site Boundary

B. RCV-018, Liquid Waste Control Valve;

Low Population Zone Boundary

C✓ PVD-6910, Liquid Effluents to Fairfield Penstocks;

Site Boundary

D. PVD-6910, Liquid Effluents to Fairfield Penstocks;

Low Population Zone Boundary

A. Plausible because the 2nd part is correct. Also plausible because RCV-018 is one of the two discharge isolation valves that may close when their respective radiation monitors sense a HI RAD.

Incorrect because RCV-018 shuts when RM-L5 senses a HI RAD, not RM-L9. The procedure that must be addressed is the RM-L9 HI RAD and TRBL Annunciator Response Procedures. Each of these procedures direct that PVD-6910 be closed, and NOT RCV-018.

B. Plausible because RCV-018 is one of the two discharge isolation valves that may close when their respective radiation monitors sense a HI RAD. The LPZ Boundary is plausible because it, along with the Site Boundary, is defined in Section 5.0 of T.S.

Incorrect because RCV-018 shuts when RM-L5 senses a HI RAD, not RM-L9. The procedure that must be addressed is the RM-L9 HI RAD and TRBL Annunciator Response Procedures. Each of these procedures direct that PVD-6910 be closed, and NOT RCV-018. Also incorrect because 2nd part is wrong - the Site Boundary is the boundary of concern.

C. CORRECT. According to GS-9, (p27, Rev 10) high radiation on RM-L9 automatically closes PVD-6910, Liquid Effluents to Fairfield Penstocks. According to ARP-019 (Rev. 1) for XCP-644, AP 2-5, the operator must verify that the automatic action occurs, which with an indication of 55 gpm still flowing, it did not. Therefore, the operator will be required to manually close PVD-6910, Liquid Effluents to Fairfield Penstocks.

According to Specification 1.1.1.1 of the Offsite Dose Calculation Manual (ODCM), the minimum number of radioactive liquid effluent monitoring instrumentation channels must be OPERABLE to ensure that the limits of ODCM Specification 1.1.2.1 are not exceeded. According to specification 1.1.2.1, this limitation provides assurance that concentrations of radioactive materials in bodies of water outside the site will not result in exposures in excess of 10CFR20 and 10CFR50 limits. According to Specification 1.1.3.1, the dose or dose commitment to an individual from radioactive materials in liquid effluents released from the site must be limited to 1.5 mrem to the total body and 5 mrem to any organ during any calendar quarter, and 3 mrem to the total body and 10 mrem to any organ during any calendar year. Therefore, the means to ensure that these dose limitations are not exceeded is to comply with the concentration limits of Specification 1.1.2.1, and the means to ensure that these concentration limitations are not exceeded is to comply with the equipment operability requirements of Specification 1.1.1.1. Since they are not being complied with (Automatic closure failure of the release flow control valve), the limits of Specification 1.1.3.1 may be exceeded.

D. Plausible because the 1st part is correct. The LPZ Boundary is plausible because it, along with the Site Boundary, is defined in Section 5.0 of T.S.

Incorrect because the 2nd part is wrong - the Site Boundary is the boundary of concern.

K/A - 068 A2.04:

Ability to (a) predict the impacts of the following on the Liquid Radwaste and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Failure of automatic isolation.

Tier: 2
Group: 2

Importance Rating: SRO 3.3

Technical References:

- ODCM 1.1.3.1

Proposed references to be provided to applicants during examination:

None

Learning Objective: HPP-710-02

Question History:

MODIFIED (Although written "from scratch", this question is similar enough to Closed Reference question LIQUID RAD WASTE 8, 19, & 42 to be classified as MODIFIED)

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

Matches the K/A in that the question tests knowledge of what automatic action should occur to terminate a liquid waste release when a high radiation condition exists.

SRO Only Justification:

SRO Only in that it tests the bases from the ODCM .

15. 073 G2.4.8 002

Given the following plant conditions:

- A Reactor Trip occurs due to a locked rotor on RCP 'B'.
- The crew has just transitioned to EOP-1.1, *Reactor Trip Recovery*.
- Annunciators RC LTDN HI RNG RM-L1 HI RAD (XCP-642, 1-5) and RC LTDN LO RNG RM-L1 HI RAD (XCP-642, 4-3) both actuate.
- A Chemistry sample indicates that Dose Equivalent I-131 activity is 1.0 microcuries/gm.

Assuming RCS activity remains at this level for another 48 hours, which ONE (1) of the following identifies the ARP and Technical Specification actions that must be taken?

A. Increase letdown to 120 gpm IAW the ARP and

be in COLD SHUTDOWN within an additional 30 hours.

B.✓ Increase letdown to 120 gpm IAW the ARP and

reduce Tavg to less than 500°F within an additional 6 hours.

C. Isolate letdown IAW the ARP and

reduce Tavg to less than 500°F within an additional 6 hours.

D. Isolate letdown IAW the ARP and

be in COLD SHUTDOWN within an additional 30 hours.

A. Plausible because the 1st part is correct - XCP-642, 1-5, Corrective Action #4 directs increasing letdown to 120 gpm. Also plausible because the requirement for COLD S/D is the same as for exceeding the transient limit in T.S. 3.4.7, Chemistry.

Incorrect since T.S. 3.4.8 requirement is HOT STANDBY w/ Tavg <500°F w/i 6 hours.

B. CORRECT. XCP-642, 1-5, Corrective Action #4 directs increasing letdown to 120 gpm. Sample results indicate that the LCO is exceeded, the Action (cooldown to less than 500°F within 6 hours) is correct.

C. Plausible because the 2nd part is correct. Also plausible because letdown is sometimes isolated, in part, to ensure habitability in the Auxiliary Building (i.e. loss AC Power). Additionally, restoring letdown is a habitability/radiological concern in many EOPs (see ERG PES Evaluations, pp 24-25). Given this concern, it seems plausible that isolating letdown would be a reasonable mitigation strategy for failed fuel.

Incorrect because the ARP requires *increasing* letdown.

D. Plausible because letdown is sometimes isolated, in part, to ensure habitability in the Auxiliary Building (i.e. loss AC Power). Additionally, restoring letdown is a habitability/radiological concern in many EOPs (see ERG PES Evaluations, pp 24-25). Given this concern, it seems plausible that isolating letdown would be a reasonable mitigation strategy for failed fuel. Also plausible because the requirement for COLD S/D is the same as for exceeding the transient limit in T.S. 3.4.7, Chemistry.

Incorrect because the ARP requires *increasing* letdown. Also incorrect since T.S. 3.4.8 requirement is HOT STANDBY w/ Tavg <500°F w/i 6 hours.

K/A - 073 G2.4.8:

(Process Radiation Monitoring) Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Tier: 2
Group: 1

Importance Rating: SRO 4.5

Technical Reference:

- ARP-019 (Rev 1), XCP-642, 4-3 (Lo Range)
- ARP-019 (Rev 1), XCP-642, 1-5 (Hi Range)
- TS LCO 3.4.8
- Step 7.3.6 of SAP-154 (Rev 1)
- Step 4.1 of EPP-001 & Attachment II, p8 of 25 (Rev 29)

Proposed references to be provided to applicants during examination:

None

Learning Objective: AB-3-25

Question History: NEW

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

The KA is matched because the operator must demonstrate knowledge of how abnormal operating procedures (i.e. ARP-019, Note that VCS uses ARPs for AOP related to the Process Rad Monitors) are used in conjunction with EOPs (EOP-1.1) with the identified rules of usage.

SRO Only Justification:

The question is SRO-Only because the operator must recall that the strategy for responding to an alarm condition on both ranges of RM-L1, the Primary Coolant Letdown Monitor and apply RCS activity levels to greater than 1-hour T.S. actions.

16. 078 A2.01 001

Given the following plant conditions:

- 100% power
- Instrument Air Compressor XAC-3B is in operation.
- Instrument Air Compressor XAC-3A is in standby.
- The Supplemental Breathing Air Compressor XAC-12 is aligned to provide Breathing Air to support on-going maintenance activities in the Aux Building.
- The Instrument Air Dryer fails, causing desiccant to clog the after filters.
- Instrument Air header pressure decreases rapidly to 35 psig.
- The crew has just entered AOP-220.1, *Loss of Instrument Air.*

Which ONE (1) of the following describes the action(s) that must be taken to restore Instrument Air header pressure AND what is the consequence of I.A. header pressure dropping below 40 psig prior to manually tripping the reactor?

A.✓ Suspend maintenance activities, open XVB-2633, IA BACKUP SYSTEM SUP HDR ISOLATION VLV, and ensure the Supplemental Breathing Air Compressor is running;

Reactor power will increase.

B. Suspend maintenance activities, open XVB-2633, and ensure the Supplemental Breathing Air Compressor is running;

Turbine blade erosion will increase.

C. Locally start the Diesel Driven Air Compressor;

Turbine blade erosion will increase.

D. Locally start the Diesel Driven Air Compressor;

Reactor power will increase.

A. CORRECT. A similar event occurred at VCS (see TB-12 (p28, Rev 12). While at 100% power, the Instrument Air Dryer failed, causing desiccant to clog the after filters. This, in turn caused a low Instrument Air header pressure alarm. The standby air compressor was started, however, Instrument Air header pressure continued to decrease, and ultimately the Turbine Group A and B Drain Valves failed open. When air pressure continued to drop, the Supplemental Air Compressor was started and air header pressure began to recover, ultimately stabilizing at 112 psig. The event established in the conditions of the question is similar, and yet different. According to TB-12 (p26, Rev 12), the Breathing Air System is usually in operation during maintenance outages, but may be required during any mode of plant operation to support maintenance requirements, as is the case here. According to TB-12 (p17), when Breathing Air is required, the Supplemental Air Compressor is aligned to supply the Breathing Air Header and XVB-2633 is closed. This means that this air compressor can NOT function as a ready backup to the Instrument Air header should an event like an After Filter clogging would require. The operator can start the Diesel Air Compressor, however, this compressor taps into the Instrument Air System upstream of the Air Dryers and therefore, will not aid in raising Instrument Air header pressure. The Only means to raise Instrument Air header pressure is to suspend maintenance activities, open XVB-2633, and ensure the Supplemental Breathing Air Compressor is running. According to ARP-001, XCP-607, 2-5, Rev 5, when IPI05875 senses less than 40 psig Group A and B Turbine Drain Valves will fail OPEN resulting in a reactor power increase.

B. Plausible because the 1st part is correct. Also plausible because, according to ARP-001, XCP-607, 2-5, Rev 5, when IPI05875 senses less than 40 psig Group A and B Turbine Drain Valves will fail OPEN. If the valves failed *CLOSED*, it is conceivable that moisture content in the steam could increase to the point where turbine blade erosion increases. Also credible because these drain valves are designed to remove moisture, which could contribute to blade erosion.

Incorrect because the failed-open valves will simply pass more steam/moisture and, if anything, will reduce the potential for turbine blade erosion.

C. Plausible because the Diesel Air Compressor is the source of I.A. after a loss of all AC power. Also plausible because AOP-220.1 (Rev 2) identifies starting this component as action to mitigate a Loss of Instrument Air. Also plausible because, according to ARP-001, XCP-607, 2-5, Rev 5, when IPI05875 senses less than 40 psig Group A and B Turbine Drain Valves will fail OPEN. If the valves failed closed, it is conceivable that moisture content in the steam could increase to the point where turbine blade erosion increases. Also credible because these drain valves are designed to remove moisture, which could contribute to blade erosion.

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Incorrect because the 1st part is wrong – The Diesel Air Compressor taps into the Instrument Air System upstream of the air dryers; therefore, it will not aid in raising Instrument Air header pressure. Also incorrect because the failed-open valves will simply pass more steam/moisture and, if anything, will reduce the potential for turbine blade erosion.

D. Plausible because the 2nd part is correct - ARP-001, XCP-607, AP2-5, Rev 5, states that when IPI05875 senses less than 40 psig, Group A and B Turbine Drain Valves will fail OPEN resulting in a reactor power increase. Plausible because the Diesel Air Compressor is the source of I.A. after a loss of all AC power. Also plausible because AOP-220.1 (Rev 2) identifies starting this component as action to mitigate a Loss of Instrument Air.

Incorrect because the 1st part is wrong – The Diesel Air Compressor taps into the Instrument Air System upstream of the air dryers; therefore, it will not aid in raising Instrument Air header pressure.

K/A - 078 A2.01:

Ability to (a) predict the impacts of the following on the Instrument Air and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Air dryer and filter malfunctions.

Tier: 2
Group: 1

Importance Rating: SRO 2.9

Technical Reference:

- AOP-220.1, Rev 2
- TB-12, p17, 26, 28, Rev 12
- ARP-001, XCP-607, AP2-5, Rev 5
- B-208-057-IA0008

Proposed references to be provided to applicants during examination:

None

Learning Objective: AOP-220.1-5

Question History: NEW

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

The KA is matched because the operator must ability to (a) predict the impacts of Air dryer and filter malfunctions on the IAS (Reactor Power will increase when Turbine Drain Valves Open); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions (realign and start the Supplemental breathing Air Compressor).

SRO Only Justification:

The question is SRO-Only because the operator must assess plant conditions during an abnormal situation (Loss of IAS) and then apply detailed knowledge of specific procedure steps; including, but not limited to: 1) Step 2 of AOP-220.1, Alternative Action, 2) SOP-220.1, Section IV.B, Step 2.1, and 3) HPP-604, Step 3.12.2, that enables recovery from the abnormal event.

17. E04 EA2.1 002

Given the following plant conditions:

- A Reactor Trip and SI have occured.
- The crew has entered EOP-1.0, *Reactor Trip/Safety Injection Actuation,* and has reached the diagnostic steps (Steps 12-14).
- RCS pressure is 1700 psig and DECREASING slowly.
- RB conditions are as follows:
    - RM-G7 and RM-G18, CNTMT HI RNG GAMMA, are NORMAL.
    - RB Sump levels are NORMAL.
    - RB pressure is 0.8 psig and STABLE.
    - Annunciators XCP-606 (607) 2-2, RBCU 1A/2A (1B/2B) DRN FLO HI, are NOT lit.
- ALL SG pressures and levels are stable.
- Auxiliary Building Area Radiation Monitors are in alarm, including the following:
    - AB VENT DUCTS RM-A11 HI RAD (XCP-645, 2-1)
    - PLANT VENT GAS RM-A3 TRBL (XCP-644, 3-4)
- NO other abnormal radiation monitoring indications exist.

Which ONE (1) of the following describes the appropriate procedure flowpath?

A. Continue in EOP-1.0, *Reactor Trip/Safety Injection Actuation;*

Transition to EOP-2.4, *Loss of Emergency Coolant Recirculation.*

B.✓ Continue in EOP-1.0, *Reactor Trip/Safety Injection Actuation;*

Transition to EOP-2.5, *LOCA Outside Containment.*

C. Go to to EOP-2.0, *Loss of Reactor or Secondary Coolant;*

Transition to EOP-2.5, *LOCA Outside Containment.*

D. Go to EOP-2.0, *Loss of Reactor or Secondary Coolant;*

Transition to EOP-2.4, *Loss of Emergency Coolant Recirculation.*

A. Plausible because the 1st part is correct. 2nd part is plausible because the given indications are consistent with a SBLOCA and EOP-2.1 would normally be used to mitigate this event. EOP-2.1 contains a Reference page transition to EOP-2.4. For these conditions, the procedure flowpath could be 1.0-2.0-2.1-2.4. EOP-2.0 also has a Reference Page transition to EOP-2.4.

Incorrect because the criteria for transition to EOP-2.0 at Step 25 of EOP-1.0 are NOT met (RCS pressure is > 250 psig)

B. CORRECT. According to EOP-1.0, conditions are NOT met for transition at the diagnostic steps (RB rad levels are normal, RB sump levels are normal, RB pressure is normal, and RBCU drain flow annunciators are not actuated, SG pressure and levels are normal and the only abnormal rad monitors are those outside the RB and not asociated with the SGs). The crew would then continue in EOP-1.0 until the transition to EOP-2.5 is met at Step 23 of EOP-1.0 (based on abnormal rad levels in the AB). The Alternate Action for Step 23 requires the crew to GO TO EOP-2.5.

C. Plausible because the 2nd part is correct. EOP-2.5 would be the next procedure used. The 1st part is plausible because there *are* indications of a loss of RCS inventory. Also plausible because there is a transition to EOP-2.0 at the diagnostic steps.

Incorrect because the criteria for transition to EOP-2.0 at Step 14 of EOP-1.0 are NOT met (RB rad levels are normal, RB sump levels are normal, RB pressure is normal, and RBCU drain flow annunciators are not actuated).

D. The 1st part is plausible because there *are* indications of a loss of RCS inventory. Also plausible because there is a transition to EOP-2.0 at the diagnostic steps. 2nd part is plausible because the given indications are consistent with a SBLOCA and EOP-2.1 would normally be used to mitigate this event. EOP-2.1 contains a Reference page transition to EOP-2.4. For these conditions, the procedure flowpath could be 1.0-2.0-2.1-2.4. EOP-2.0 also has a Reference Page transition to EOP-2.4.

Incorrect because the criteria for transition to EOP-2.0 at Step 14 of EOP-1.0 are NOT met (RB rad levels are normal, RB sump levels are normal, RB pressure is normal, and RBCU drain flow annunciators are not actuated).

K/A - E04 EA2.1:

Ability to determine and interpret the following as they apply to the (LOCA Outside Containment). Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Tier: 1
Group: 1

Importance Rating: SRO 4.3

Technical Reference:

- EOP-2.0, Rev 13
- EOP-2.4, Rev 11
- EOP-2.5, Rev 7

Proposed references to be provided to applicants during examination:

None

Learning Objective: EOP-1.0-07

Question History:

MODIFIED (Although written "from scratch", this question is similar enough to Closed Reference questions EOPS 439, 459, 471, 478, & 525 to be classified as MODIFIED)

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

The KA is matched because the operator must demonstrate the ability to interpret a given set of plant conditions and select an appropriate procedure flowpath associated with an unisolable LOCA outside the Containment.

SRO Only Justification:

The question is SRO-Only because the question involves assessing plant conditions of an abnormal/emergency event, and then prescribing a procedure, and procedure flowpath (including a transition to an ES procedure, which is NOT RO knowledge).

18. E08 G2.2.25 003

When implementing EOP-16.0, *Response to Imminent Pressurized Thermal Shock*, which ONE (1) of the following identifies the required soak time and the bases for the soak?

A.✓ ONE (1) hour;

prevent flaw growth in the Reactor Vessel wall

B. ONE (1) hour;

relieve compressive stress on the inner wall of the Reactor Vessel

C. EIGHT (8) hours;

prevent flaw growth in the Reactor Vessel wall

D. EIGHT (8) hours;

relieve compressive stress on the inner wall of the Reactor Vessel

A. CORRECT: IAW EOP-16.0, Step 24. According to HFRP1BG, page 2, the objective of EOP-16.0 is to prevent the growth of a flaw.

B. Plausible because the 1st part is correct IAW EOP-16.0, Step 24. Also plausible because compressive stress is a stress mechanism on the *outer* wall due to thermal stresses.

Incorrect because both pressure and thermal stresses on a cooldown are tensile on the inner wall (see Figure TS15.24).

C. Plausible because the 2nd part is correct. 1st part plausible because this is a soak period required for PZR safety valves during a plant heatup.

Incorrect because the soak period is only 1 hour per Step 24 of EOP-16.0.

D. Plausible because this is a soak period required for PZR safety valves during a plant heatup. Also plausible because compressive stress is a stress mechanism on the *outer* wall due to thermal stresses.

Incorrect because the soak period is only 1 hour per Step 24 of EOP-16.0. Also incorrect because both pressure and thermal stresses on a cooldown are tensile on the inner wall (see Figure TS15.24).

K/A - E08 G2.2.25:

(Pressurized Thermal Shock) Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Tier: 3
Group: 1

Importance Rating: SRO 4.2

Technical Reference:

- T.S. 3.4.9.1
- T.S. Figures 3.4-2 & 3.4-3
- T.S. Bases 3/4.4.9.1)b)
- EOP-16.0, Step 24

Proposed references to be provided to applicants during examination:

None

Learning Objective: EOP-16.0-07

Question History: NEW

10 CFR Part 55 Content: 43(b)(2)

K/A Match:

Meets K/A by application of TS Pressure-Temperature Limitations.

SRO Only Justification:

SRO-level because it requires detailed knowledge of EOP-16.0 and the bases for a step that is not an Immediate Operator Action.

19. G2.1.27 003

Which ONE (1) of the following describes the basis for OPERABILITY limits on accumulator volume and boron to ensure that assumptions used for accumulator injection in the safety analysis are met?

ONLY_______accumulator(s) is(are) required to provide the initial core cooling for a postulated Loss of Coolant Accident and limit the maximum power that may be reached during _______.

A. ONE

large secondary pipe ruptures

B. TWO

an over feeding event as a result of a feedwater malfunction

C. ONE

an over feeding event as a result of a feedwater malfunction

D.✓ TWO

large secondary pipe ruptures

A. Plausible because ONLY ONE (1) train of safeguards equipment is required to mitigate any accident. Also plausible because the 2nd half is correct. (Ref. T.S. Bases 3/4.5.1)

Incorrect because, according to VCSNS FSAR (Table 15.4-5), presents the reflood mass and energy release to the containment and the *broken* loop accumulator mass and energy flowrate to containment. i.e. it is postulated that the broken loop accumulator mass is unavailable for reflood.

B. Plausible because the 1st part is correct. During a Large Break Loss of Coolant accident as described in Chapter 15.4 of the FSAR (Ref. 15.4.1.1.2 VCSNS FSAR) when the RCS depressurizes to 600 psia the accumulators begin to inject water into the reactor coolant loops. From the latter stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for reflooding of the reactor core. However, according to VCSNS FSAR (Table 15.4-5) presents the reflood mass and energy release to the containment and the *broken* loop accumulator mass and energy flowrate to containment. i.e. it is postulated that the broken loop accumulator mass is unavailable for reflood. 2nd part is plausible because it is a significant heat removal transient as described in, (Ref 15.2.10 of VCSNS FSAR; Excessive

Heat Removal Due To Feedwater Sysytem Malfunction), which would result in an increase in reactor power for which if accumulators could inject would terminate the power rise.

Incorrect because the T.S. bases for the accumulators states specifically the borated water serves to limit the maximum power which may be reached during large secondary pipe ruptures.

C. Plausible because ONLY ONE (1) train of safeguards equipment is required to mitigate any accident. 2nd part is also plausible because it is a significant heat removal transient as described in, (Ref 15.2.10 of VCSNS FSAR; Excessive Heat Removal Due To Feedwater Sysytem Malfunction), which would result in an increase in reactor power for which if accumulators could inject would terminate the power rise.

Incorrect because, according to VCSNS FSAR (Table 15.4-5) presents the reflood mass and energy release to the containment and the *broken* loop accumulator mass and energy flowrate to containment. i.e. it is postulated that the broken loop accumulator mass is unavailable for reflood. Also incorrect because the T.S. bases for the accumulators states specifically the borated water serves to limit the maximum power which may be reached during large secondary pipe ruptures.

D. CORRECT: During a Large Break Loss of Coolant accident as described in Chapter 15.4 of the FSAR (Ref. 15.4.1.1.2 VCSNS FSAR) when the RCS depressurizes to 600 psia the accumulators begin to inject water into the reactor coolant loops. From the latter stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for reflooding of the reactor core. However, according to VCSNS FSAR (Table 15.4-5) presents the reflood mass and energy release to the containment and the *broken* loop accumulator mass and energy flowrate to containment. i.e. it is postulated that the broken loop accumulator mass is unavailable for reflood. Also correct because the T.S. bases for the accumulators states specifically the borated water serves to limit the maximum power which may be reached during large secondary pipe ruptures.

K/A - G2.1.27:

(Conduct of Operations) Knowledge of system purpose and/or function.

Tier: 3
Group: 1

Importance Rating: SRO 4.0

Technical Reference:

- Chapter 15.4.1.1.2 VCSNS FSAR
- TS Bases 3/4.5.1
- Chapter 15.2.10 of VCSNS FSAR

Proposed references to be provided to applicants during examination:

None

Learning Objective: AB-10-21

Question History: NEW

10 CFR Part 55 Content: 43(b)(2)

K/A Match:

Meets K/A by application of TS Bases which describes the function/purpose of the ECCS Accumulators.

SRO Only Justification:

SRO-level because it requires detailed knowledge of a GOP and the bases for a T.S. that is NOT a Safety Limit.

20. G2.1.34 001

Which ONE (1) of the choices below completes the following statement?

In accordance with GOP-2, *Plant Startup and Heatup (Mode 5 to Mode 3)*, Dissolved Oxygen must be within limits prior to exceeding _____.

The basis for this limit, per Technical Specification 3.4.7, is to prevent ________ corrosion.

A.✓ 200°F;

stress

B. 200°F;

crevice

C. 350°F;

stress

D. 350°F;

crevice

A. CORRECT. GOP-2, Reference Page, Item 2.E, requires RCS chemistry to be in-spec prior to exceeding 200°F. Step 3.1 also contains this requirement. Additionally, CP-614, Attachment III, requires Oxygen to be < 0.1 ppm prior to exceeding 200°F.

B. Plausible because the 1st part is correct. 2nd part is plausible because crevice corrosion is one form of corrosion and *is* associated with oxygen.

Incorrect because T.S. bases for LCO 3.4.7 specifically states that RCS Chemistry limits are designed to minimize and reduce the potential for *stress* corrosion.

C. Plausible because the 2nd part is correct. Also plausible because MODE 4 ends at 350°F and is transition through in GOP-2.

Incorrect because GOP-2 is more restrictive in that Oxygen must be within limits prior to exceeding 200°F.

D. Plausible because the note in T.S. Table 3.4-2 states "*Limit is not applicable with Tavg < 250°F." 2nd part is plausible because crevice corrosion is one form of corrosion and *is* associated with oxygen. Also plausible because MODE 4 ends at 350°F and is transition through in GOP-2.

Incorrect because T.S. bases for LCO 3.4.7 specifically states that RCS Chemistry limits are designed to minimize and reduce the potential for *stress* corrosion. Also incorrect because GOP-2 is more restrictive in that Oxygen must be within limits prior to exceeding 200°F.

K/A - G2.1.24:

(Conduct of Operations) Knowledge of primary and secondary chemistry limits.

Tier: 3
Group: 1

Importance Rating: SRO 3.5

Technical Reference:

- T.S. Table 3.4-2 (Page 255 of 588)
- T.S. 3.4.7 (Page 254 of 588)

Proposed references to be provided to applicants during examination:

None

Learning Objective: SB-4-15

Question History:

MODIFIED (Although written "from scratch", this question is similar enough to Open Reference question NORMAL OPERATIONS 58 to be classified as MODIFIED)

10 CFR Part 55 Content: 43(b)(2),(5)

K/A Match:

Meets K/A by application of TS RCS Oxygen Chemistry requirements.

SRO Only Justification:

SRO-level because it requires detailed knowledge of a GOP and the bases for a T.S. that is NOT a Safety Limit.

21. G2.2.38 002

Given the following plant conditions:

- The unit was at 100% power when a high pressure feedwater heater bypass valve failure caused Reactor power to peak at 2910 MWt on U9002.
- The failed valve has been closed and Reactor power is at 100%.

Which ONE (1) of the choices below completes the following statement regarding follow-up actions for these conditions?

In accordance with OAP-100.6, the Reactor power limit of 100% is based on ______ .

Notify the NRC within ______.

REFERENCE PROVIDED

A. a five-minute rolling average power;

1 hour.

B. a five-minute rolling average power;

24 hours.

C. an integrated shift average power;

1 hour.

D✓ an integrated shift average power;

24 hours.

A. Plausible because the *102%* overpower is evaluated by checking the five minutes rolling average computer point. 2nd part is plausible because a one hour report would be required (per T.S. 6.7, Safety Limit Violation) if Core Power Safety Limit was violated.

Incorrect because the 2900 MWt limit is based on an 8-hour rolling average. Also incorrect because the report is a 24 hour requirement, not one hour.

B. Plausible because the 2nd part (reporting requirement) is correct. Also plausible because the *102%* overpower is evaluated by checking the five minutes rolling average computer point.

Incorrect because the 2900 MWt limit is based on an 8-hour rolling average.

C. Plausible because the 1st part is correct. 2nd part is plausible because a one hour report would be required if Core Power Safety Limit was violated.

Incorrect because the report is a 24 hour requirement, not one hour.

D. CORRECT. The action is correct because licensed power limit compliance is maintained by monitoring an 8 hour shift average per OAP-100.6, Section 10.5.a.. A 24 hour report for exceeding the license limit is required by OAP-100.06 (Section 10.5); NL-122, REGULATORY NOTIFICATION AND REPORTING (Enc. A, Item P-9).

10.0 GUIDELINES FOR OPERATION AT THE LICENSED LIMIT

10.1 Display U9002, EIGHT-HOUR SHIFT AVG REACTOR PWR, on one of the Analog Trend Recorders.

10.2 Ensure both of the following are displayed in the Control Room:

a. U9002, SHIFT AVG POWER.

b. U9003, QCORE1 (C1M).

10.3 Adjust core power to obtain an integrated shift average value between 2895 and 2900 MWt.

NOTE 10.4

Refer to Enclosure F, Notes on Q-Core and I&C Op Tests, for a discussion of the Q-Core calculation.

10.4 If the IPCS is not available, QCORE 1 or U9002, SHIFT AVERAGE POWER is invalid, then perform the following:

a. Monitor the following on the NIS power range drawers:

1) N41A, PERCENT FULL POWER.
2) N42A, PERCENT FULL POWER.
3) N43A, PERCENT FULL POWER.
4) N44A, PERCENT FULL POWER.

b. Adjust Reactor power as necessary to maintain the highest reading NI at less than or equal to 100%.

10.5 If actual power exceeds either of the following limits, initiate a 24-hour report to the NRC per NL-122, Regulatory Notification And Reporting and take prompt action to reduce power to less than 2900 MWt:

a. 2900 MWt (100%) as integrated over each eight-hour shift per computer point U9002.

b. 2958 MWt (102%) at any time, as determined by analysis of power indications (i.e. U9003-5M, average NI power indication, delta-T, etc).

10.6 If the computer alarm (U9003) is received for reference core power exceeding 2929 MWt (101%), perform the following:

a. Verify the alarm is valid by monitoring the five-minute rolling average (U9003-5M).

b. If the alarm is determined to be valid as indicated by U9003-5M greater that 2929 MWt (101%), initiate a CR to evaluate the transient and take prompt action to reduce power to less than 2900 MWt.

c. If the alarm is determined to be the result of a feedwater flow transient or other calculated inputs to U9003, which are not related to actual core power, make a station log entry denoting the reason for the alarm.

NOTE 10.7

L0303S is the previous 20 one-minute consecutive calculations for U9003 (alarm only), indicating that reference core power has exceeded 2914.5 MWt (100.5%) for greater than twenty minutes

10.7 If the computer alarm is received for L0303S:

a. Verify the alarm is valid.
b. Take prompt action to reduce power to less than 2900 MWt.
c. Write a CR to evaluate the transient.

NOTE 11.0

These expectations are in addition to those illustrated in SAP-102, Statement Of Responsibilities, Operations, SAP-155, Reactivity Management, SAP-200, Conduct Of Operations, and Management Directive-57, Reactivity Management.

For newly refueled cores refer to Enclosure G, Hot Leg Streaming, for an explanation of anticipated rod motion.

K/A - G2.2.38:

(Equipment Control) Knowledge of conditions and limitations in the facility license.

Tier: 3
Group: 2

Importance Rating: SRO 4.5

Technical Reference:

- OAP-100.6, Section 10.0 (Pages 11 and 12)
- NL-122, Enclosure A, (pp 1 through 20)
- VCSNS Operating License, Section 2.C.(1)

Proposed references to be provided to applicants during examination:

- **NL-122 (only pp 1-20 of Enclosure A)**

Learning Objective: OAP-100.6-7

Question History: NEW

10 CFR Part 55 Content: 43(b)(1)

K/A Match:

Meets K/A by requiring knowledge of action(s) associated with exceeding the reactor power limits specified in the facility license.

SRO Only Justification:

SRO only because it requires the direction of compensatory actions and knowledge of reporting requirements. Because of the importance, the license limit and actions may be known by RO and SRO candidates but the direction of corrective actions and internal/external reporting are the responsibility of the Shift Supervisor (see NL-122, Sections 6.1 and 6.2).

22. G2.3.13 001

Given the following plant conditions:

- 100% power
- A QA Audit has determined that an ECCS valve lineup was not properly documented and may be incorrect.
- A Reactor Building entry has been approved to verify the position of the valves in question.

Which ONE (1) of the choices below completes the following statements?

The team will normally enter the Reactor Building through the ______ Airlock. At the completion of this entry, per OAP-108.1, *Control of Reactor Building Entry,* the operating crew must generate an R&R to ensure that ________ is conducted.

A. ESCAPE;

STP-109.001, *Reactor Building Closeout Inspection,*

B.✓ ESCAPE;

STP-215.001B, *Reactor Building Personnel Escape Airlock Test,*

C. PERSONNEL;

STP-109.001, *Reactor Building Closeout Inspection,*

D. PERSONNEL;

STP-215.001A, *Reactor Building Personnel Airlock Test,*

A. Plausible because the airlock is correct and the closeout inspection is required at the completion of the entry.

Incorrect because, per OAP-108.1, Section 6.5.b., the Action R&R is intended to ensure STP-215B is completed within 7 days, not to ensure STP-109.001 is conducted.

B. CORRECT. Per OAP-108.01, *Control of Reactor Building Entry*, the ESCAPE Airlock is used to reduce exposure to neutron streaming. Per OAP-108.1, Section 6.5.b., the Action R&R is intended to ensure STP-215B is completed within 7 days

C. Plausible because this is the normal entry point for non-power entries. Also plausible because STP-109.001 must be performed at the completion of the entry.

Incorrect because, per OAP-108.1, Section 6.5.b., the Action R&R is intended to ensure STP-215B is completed within 7 days, not to ensure STP-109.001 is conducted. Also incorrect because the ESCAPE hatch is used to reduce exposure to neutron streaming.

D. Plausible because this is the normal entry point for non-power entries. Also plausible because, if the PERSONEL AIRLOCK were used, this would be the correct STP.

Incorrect because the ESCAPE hatch is used, not the PERSONEL AIRLOCK.

K/A - G2.3.13:

(Radiation Control) Knowledge of radiological safety procedures pertaining to licensed operator duties.

Tier: 3
Group: 3

Importance Rating: SRO 3.8

Technical Reference:

- OAP-108.1, 6.4/6.5 (Pages 3 and 4 of 5)

Proposed references to be provided to applicants during examination:

None

Learning Objective: OAP-108.1-03

Question History: NEW

10 CFR Part 55 Content: 43(b)(4), (5)

K/A Match:

Meets K/A by requiring detailed knowledge of the administrative requirements (radiological safety practices - use Escape Hatch to reduce exposure to neutron streaming) associated with the RB entry procedure (OAP-108.1).

SRO Only Justification:

SRO only in that the Duty Shift Supervisor implements OAP-108.1 and the CRS must direct the operating crew to initiate the R&R to ensure completion of STP-215B.

23. G2.3.14 005

Given the following plant conditions:

- A Reactor Trip and SI have occurred due to a LOCA.
- The crew has implemented EOP-2.0, *Loss of Reactor or Secondary Coolant.*

Which ONE (1) of the choices below completes the following statement?

Control Room Emergency Ventilation must be reduced to one train in operation within __________ to limit the dose to Control Room personnel to a maximum of __________ whole body?

A. TWO (2) hours;

5.0 rem

B. TWO (2) hours;

1.0 rem

C.✓ THIRTY (30) minutes;

5.0 rem

D. THIRTY (30) minutes

1.0 rem

A. Plausible because the 2nd part is correct. Also, 2 hours is plausible because this is the minimum time frame for RB Spray operation, as addressed in EOP-2.0, NOTE - Step 11.d.

Incorrect because according to the Reference Page for EOP-2.0, CR ventilation must be reduced to one train within 30 minutes, not 2 hours.

B. 2 hours is plausible because this is the minimum time frame for RB Spray operation, as addressed in EOP-2.0, NOTE - Step 11.d. Also plausible because 1.0 R is the TEDE administrative limit at VCSNS. This section relates to the need to reduce CR ventilation to 1 train (to limit outside air intake to 1000 scfm) within 30 minutes of a DBA LOCA.

Incorrect because according to the Reference Page for EOP-2.0, CR ventilation must be reduced to one train within 30 minutes, not 2 hours. Also incorrect since T.S. Bases 3/4.7.6, Control Room Normal and Emergency Air Handling System indicates that the limiting dose during accident conditions is 5.0 R whole body, not 1.0 R.

C. CORRECT. T.S. Bases 3/4.7.6, Control Room Normal and Emergency Air

Handling System indicates that the limiting dose during accident conditions is 5.0 R whole body. The link between the EOP requirement to reduce ventilation to 1 train and the T.S. bases is found in 10CFR50, Appendix A, GDC #19, which is excerpted below. The link is further established in Section 15.4.1.4.4.6, which is also exerpted below.

D. Plausible because the 1st part is correct. Also plausible because 1.0 R is the TEDE administrative limit at VCSNS.

Incorrect since T.S. Bases 3/4.7.6, Control Room Normal and Emergency Air Handling System indicates that the limiting dose during accident conditions is 5.0 R whole body, not 1.0 R.

<u>10CFR50, Appendix A, General Design Criteria 19:</u>

Criterion 19--Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

<u>FSAR Chapter 15, Section 15.4.1.4.4.6 Excerpt:</u>

For analysis purposes, both trains of the Control Room Ventilation System are assumed to start in the emergency mode. Upon actuation, an isolation damper in one train is also assumed to fail open. This leads to a period of high unfiltered inleakage into the control room until operator action is credited. In accordance with the Emergency Operating Procedures, the operator is required to manually shutoff the affected train within 30 minutes, thereby, terminating this inleakage path. Thereafter, only one train of the Control Room Ventilation System is assumed to be operating.

For the purpose of the Regulatory Guide 1.4 analysis, the maximum allowable air intake value was determined for the limiting total integrated dose (i.e., 30 Rem thyroid) to control room personnel. The results are presented in Table 15.4-18. The maximum allowable air intake value was determined to be 2100 cfm. The system operating flow of 1000 cfm provides adequate margin for the protection of control room personnel as specified under General Design Criterion 19 of 10CFR50, Appendix A. The maximum allowable value for other sources of unfiltered inleakage was also determined with air intake set at the Technical Specification limit. For the limiting total integrated dose (i.e., 30 Rem thyroid), the maximum allowable value of unidentified unfiltered inleakage was determined to be 55 cfm per operating train.

K/A - G2.3.14:

(Radiation Control) Knowledge of radiological or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Tier: 3
Group: 3

Importance Rating: SRO 3.8

Technical Reference:

- T.S. 3.11.2.6 (Page 453 of 588)
- T.S. Basis 3.11.2.6 (Page 544 of 588)

Proposed references to be provided to applicants during examination:

None

Learning Objective: EOP-2.0-04; GS-8-18

Question History: NEW

10 CFR Part 55 Content: 43(b)(2), (4)

K/A Match:

Meets K/A by requiring knowledge of actions to take (reduce CR ventilation) for radiological hazards (limiting 10CFR50, App. A, GDC 19 dose) to Control Room personnel during emergency conditions (LOCA).

SRO Only Justification:

SRO only because it requires knowledge of T.S. bases (3/4.7.6) that is not a Safety Limit.

24. G2.4.20 001

Given the following plant conditions:

- Component Cooling Water is in a normal alignment with the following exception:
  - (CCW) Pump 'C' is not available.
- CCW Loop 'A' is the Active Loop.
- A large break LOCA has occurred.
- The operating crew is performing EOP-2.0, *Loss of Reactor or Secondary Coolant*, Step 16 – Verify equipment is available for Cold Leg Recirculation.

Which ONE (1) of the following is the REQUIRED action with respect to shifting CCW to FAST speed?

A. Shift CCW Pump 'A' to FAST and continue in EOP-2.0.

B.✓ Leave CCW Pump 'A' in SLOW and continue in EOP-2.0.

C. Shift CCW Pump 'B' to FAST speed, swap active loops per SOP-118, *Component Cooling Water System*, THEN continue in EOP-2.0.

D. Reduce CCW loads per EOP-2.2, *Transfer to Cold Leg Recirculation*, THEN continue in EOP-2.0.

A. Plausible because this is a potential alignment in SOP-118 if CCW Pump 'C' was running. Also plausible because it would accomplish the desired end state. Additionally, "continue in EOP-2.0" is plausible because it is correct (see CAUTION - Step 16.c, 2nd bullet).

Incorrect because CAUTION - Step 16.c prohibits shifting the running pump to FAST if the sing CCW pump is not available, as in the given conditions.

B. CORRECT. EOP-2.0, CAUTION - Step 16.c prohibits shifting the running pump to FAST if the sing CCW pump is not available, as in the given conditions. "Continue in EOP-2.0" is correct (see CAUTION - Step 16.c, 2nd bullet).

C. Plausible because this is a possible solution and because it is an evolution addressed in SOP-118. Also plausible because it may ultimately be the correct alignment when time allows in another EOP.

Incorrect because CAUTION - Step 16.c gives specific direction NOT to realign anything at this time and continue in EOP-2.0.

D. Plausible because this *will* be accomplished in EOP-2.2 per the 2nd bullet of CAUTION - Step 16.c.

Incorrect because it will NOT be accomplished until the crew implements EOP-2.2.

K/A - G2.4.20:

(Emergency Procedures/Plans) Knowledge of the operational implications of EOP warnings cautions and notes.

Tier: 3
Group: 4

Importance Rating: SRO 4.3

Technical Reference:

- EOP-2.0, Step 16.c (Page 10 of 15)
- SOP-118, Section IV.D (Pages 20 and 21)

Proposed references to be provided to applicants during examination:

None

Learning Objective: EOP-2.0-05, 06

Question History: NEW

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

Meets K/A by requiring an operational decision based on an EOP-2.0 NOTE and knowledge of the SOP (SOP-118).

SRO Only Justification:

SRO only because it requires application of ***specific*** mitigating strategies in EOP-2.0. Selection of procedures per 10CFR55.43(b)(5) is also involved in the process of eliminating choices C & D and in selecting the correct answer.

25. G2.4.41 003

Which ONE (1) of the following would REQUIRE a declaration of a Site Area Emergency per EPP-001, *Activation and Implementation of the Emergency Plan*?

The event has been in progress for TWENTY (20) minutes.

A. loss of ONLY the following:

1) AC power capability to 7.2 KV ESF buses 1DA and 1DB reduced to a single power source
2) < 108 VDC on both Train A and Train B vital 125VDC systems

B. loss of ONLY the following:

1) 115 KV power to XTF-4 and XTF-5
2) 230 KV power to XTF-31
3) Parr Hydro Plant 13.8 KV power to ESF Bus 1DA or 1DB

C. ALL of the following:

1) AC power capability to 7.2 KV ESF buses 1DA and 1DB reduced to a single power source
2) Any additional single power source failure will result in loss of all AC power to both 7.2 KV ESF buses.

D✓ loss of ALL of the following:

1) 115 KV power to XTF-4 and XTF-5
2) 230 KV power to XTF-31
3) Parr Hydro Plant 13.8 KV power to ESF Bus 1DA or 1DB
4) Diesel Generator A
5) Diesel Generator B

A. Plausible because the given 10 minutes exceeds the threshold for an ALERT. Also plausible because 2nd part contains the DC Detection Methods for a SAE and part of the AC Detection methods(see EPP-001, Att. II, Page 12 of 25).

Incorrect because 10 minutes does not exceed the SAE threshold of 15 minutes. Also incorrect because all of the Detection Methods are NOT met.

B. Plausible because the 1st part is correct - the given 20 minutes exceeds the threshold for a SAE.

Incorrect because 2nd part contains only 3 of 5 Detection Methods for a SAE.

C. Plausible because the given 10 minutes exceeds the threshold for an ALERT. Also plausible because 2nd part contains Detection Methods for an ALERT and would be correct if the given time was >15 minutes.

Incorrect because 10 minutes does not exceed the SAE threshold of 15 minutes.

D. CORRECT. Per EPP-001, Att. II, Page 11 of 25, an SAE is required when ALL of the Detection Methods are met for more than 15 minutes.

K/A - G2.4.41:

(Emergency Procedures/Plans) Knowledge of the emergency action level thresholds and classifications.

Tier: 3
Group: 4

Importance Rating: SRO 4.6

Technical Reference:

- EPP-001, Attachment II, Page 1 (Pages 11 & 12 of 25)

Proposed references to be provided to applicants during examination:

None

Learning Objective: EPP-001-4095

Question History: NEW

10 CFR Part 55 Content: 43(b)(5)

K/A Match:

Meets K/A by requiring knowledge of Detection Methods for declaration of a SAE and an Alert.

SRO Only Justification:

SRO Only because knowledge of the EAL Tables is required.