1. 0026G2.1.7 1
Unit 1 Initial Conditions:

- The operations team is cooling down the unit in preparation for refueling in accordance with 1-GOP-2.6, "UNIT COOLDOWN, LESS THAN 205 °F TO AMBIENT."
- The pressurizer (PRZR) is water-solid, and all PRZR heaters are tagged off.
- RCS Pressure is approximately 250 psig.
- RCS Temperature is approximately 180 °F.
- 'A' and 'B' S/G WR levels are approximately 98%. 'C' S/G NR level is 65%.
- All RCPs are stopped.

Current conditions:

- A large unisolable CCW leak caused a complete and sustained loss of CCW.
- The operations team entered 1-AP-27.00, "LOSS OF DECAY HEAT REMOVAL CAPABILITY."
- The operators were UNABLE to control RCS temperature using natural circulation cooling.
- CETC temperatures are approaching saturation.

Based on the current conditions, which one of the following is the NEXT method of providing decay heat removal, in accordance with AP-27.00?

A. Forced feed cooling.

B. Reflux boiling heat removal.

C. Gravity feed cooling.

D. Cooling the RCS with the SFP and RWST coolers.

K/A

Loss of Component Cooling Water: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
(CFR: 41.5/43.5/45.12/45.13) (SRO - 4.7)

K/A Match Analysis

Given a complete loss of component cooling water under S/D and C/D conditions, the applicant must use the plant conditions to determine the appropriate course of action.

SRO-Only Analysis

See attached SRO-only guidance flowchart. As an amplification, this question is focusing on the correct procedural selection of the various attachments in AP-27.00 (the four answer choices are word-for-word the titles of the various attachments in AP-27.00); and is therefore testing procedural knowledge on a different and more detailed level than what is expected for a RO.

Answer Choice Analysis

A. INCORRECT. Attachment 4 of AP-27.00 requires a transition to Attachment 5 to establish reflux boiling heat transfer for the given condition. Plausible because 1-OSP-ZZ-004 specifies that forced feed and bleed cooling is a possible "mandatory backup cooling method" in the initial given plant conditions.

B. CORRECT. Attachment 4 of AP-27.00 requires a transition to Attachment 5 to establish reflux boiling heat transfer for this condition.

C. INCORRECT. See analysis of A. above. Plausible because gravity feed cooling is a method specified as attachment 8 of AP-27.00.

D. INCORRECT. See comments for A. above. Plausible because cooling the RCS with the SFP and RWST coolers is a cooling method as specified in attachment 10 of AP-27.00.

Supporting References

- 1-GOP-2.6, "UNIT COOLDOWN, LESS THAN 205 F TO AMBIENT," rev 28 (p. 8, 12, 18, 19, 20-22)

-SPS TS Fig. 3.1-2, "RCS COOLDOWN LIMITATIONS."

-1-AP-15.00, "LOSS OF COMPONENT COOLING," CAUTION before step 1.

1-AP-27.00, "LOSS OF DECAY HEAT REMOVAL CAPABILITY," rev 18; procedural flowpath to steps 19, 20, and 21; attachments 4, 5, 6

1-OSP-ZZ-004, "UNIT 1 SAFETY SYSTEMS STATUS LIST FOR COLD SHUTDOWN/REFUELING CONDITIONS," rev 35, p. 10 (table of mandatory and non-mandatory backup cooling methods)

References Provided to Applicant
none

Answer: B

2. 0036AA2.03 1

In accordance with the Surry Power Station FSAR Accident Analysis, which one of the following Fuel Handling Accident conditions result in a HIGHER total effective dose equivalent (TEDE) received at the Exclusion Area Boundary (EAB) than what is assumed in the accident analysis?

Consider that ALL OTHER assumptions and conservatisms inherent in the analysis remain UNCHANGED, except for the individual condition below.

A. The delay time from reactor shutdown to the initiation of fuel assembly transfer operations is 148 hours.

B. The analysis of a postulated fuel handling accident in containment is based on 50% of the fuel assembly Iodine-131 activity assumed to be released into the reactor cavity water.

C. The total activity released from a fuel handling accident in containment is assumed to be released instantaneously.

D. The analysis of a postulated fuel handling accident in the spent fuel pool is based on a fuel radionuclide inventory derived from a rated core power level of 2546 MWt.

K/A

Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Magnitude of potential radioactive release.
(CFR: 43.5/45.13) (SRO - 4.2)

K/A Match Analysis

The question requires the applicant to understand the assumptions that are behind the fuel handling accident (FHA) analysis as presented in the Surry FSAR.

SRO-Only Analysis

The applicant is required to know and understand the severity factors inherent in the FSAR/design basis accidents for fuel handling that are outside the knowledge requirement for ROs.

Answer Choice Analysis

A. INCORRECT. On page 14.4-6 and 14.4-8 of the UFSAR, the accident analyses assume "a delay time from reactor shutdown to the initiation of fuel assembly transfer operations is at least 100 hours." Furthermore, Surry Technical Specification 3.10

requires a minimum 100-hour period between the shutdown of a unit and initiation of fuel movement. Therefore, the wording and the exactitude of the number's specification (100 hours plus two days) is plausible. However, the distractor is incorrect, because a delay time that is longer than the 100 hrs assumed in the analysis will result in a LOWER dose, NOT a HIGHER dose as required by the question stem.

B. CORRECT ANSWER. As specified in the UFSAR page 14.4-6, "9. 5.35 percent of the fuel assembly Iodine-131 activity is assumed to be released into the reactor cavity water, as are five percent of the other iodine isotopes present in the fuel assembly, 99.85% being elemental and 0.15% in the organic form. The decontamination factor (DF) for elemental ioding is 500 while the DF for organic iodine is 1." The correct answer is plausible because 50% of the iodine activity is a plausible design criteria, but much greater than what is actually assumed in the accident analysis.

C. INCORRECT. The Surry UFSAR states on p. 14.4-6, that for a fuel handling accident in containment, "More specific conservative assumptions are: 1. A puff release of radioactivity occurs as the result of the rupture of a fuel assembly in the reactor fuel cavity. The puff relase is instantaneously and uniformly distributed through one-half the containment volume." Therefore, answer "C" is plausible because it is an actual assumption used in the analysis. To further add to the plausibility, if the analysis had assumed a certain finite release time, changing this parameter to model the accident as an instantaneous release would result in a higher dose--which is what the question stem is asking for. The distractor is incorrect because it is an assumption in the analysis, and does not, in fact, result in a HIGHER dose.

D. INCORRECT. The distractor is derived from one of the actual assumptions used in the analysis. Page 14.4-8 of the UFSAR states, "The fuel radionuclide inventory was based on a core power level of 2605 MWt. This core power level is conservative compared to 102% of the uprated power level of 2546 MWt (i.e., 2597 MWt)." Therefore, answer "D" is plausible because it uses language from the actual assumption used in the analysis. The distractor is incorrect because it states a lower power level than what is assumed in the analysis, and therefore does not, in fact, result in a HIGHER dose.

Supporting References

-Surry Power Station UFSAR rev 36 section 14.4.1, "Fuel-Handling Accidents."

-Surry Power Station Technical Specifications 1.0 (p. 1.0-1) and 3.10 (p. 3.10-3 and p. 3.10-9).

-The question developer constructed this question by modifying a similar question found in an Indian Point unit 2 ILO exam given in 2005.

References Provided to Applicant
none

Answer: B
3. 0039A2.03 1
Unit 1 Initial Conditions:

- 100% Power
- A tube leak in the 'B' Steam Generator (S/G) has been identified.
- Control room operators have transitioned to 1-AP-24.00, "MINOR SG TUBE LEAK."

Current conditions:

- Condenser air ejector radiation monitor, RI-SV-111, alarms but the automatic actions do NOT occur.
- Main Steam (MS) Line B radiation monitor, RI-MS-125, alarms.
- MS Line A and C radiation monitor readings are slightly higher than before.
- The Senior Reactor Operator directs a manual reactor trip and initiation of 1-E-0, "REACTOR TRIP OR SAFETY INJECTION."
- Safety Injection (SI) does NOT automatically actuate.
- At step 4 of 1-E-0, it is determined that SI is NOT REQUIRED.

Based on the current conditions, which one of the following is (1) the correct procedural flowpath, AND (2) the correct method to procedurally address the failure of RV-SI-111 automatic actions?

A. (1) Transition to 1-ES-0.1, "REACTOR TRIP RESPONSE."
(2) Perform steps in 1-AP-24.00 to correct the failure of RV-SI-111 automatic actions, in parallel with 1-ES-0.1.

B. (1) Transition to 1-ES-0.1, "REACTOR TRIP RESPONSE."
(2) Perform steps in 1-AP-24.01 to correct the failure of RV-SI-111 automatic actions, in parallel with 1-ES-0.1.

C. (1) Transition to 1-AP-24.01, "LARGE STEAM GENERATOR TUBE LEAK."
(2) Perform steps in 1-AP-24.01 to correct the failure of RV-SI-111 automatic actions.

D. (1) Transition to 1-AP-24.01, "LARGE STEAM GENERATOR TUBE LEAK."
(2) Perform steps in 1-AP-24.00 to correct the failure of RV-SI-111 automatic actions, in parallel with 1-AP-24.01.

K/A

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

consequences of those malfunctions or operations:
Indications and alarms for main steam and area radiation monitors (during SGTR).
(CFR: 41.5/43.5/45.3/45.13) (SRO - 3.7)

K/A Match Analysis

Requires the applicant to identify the situation, given a set of conditions, and exercise the correct procedures to mitigate both the SGTR and a failure of SJAE radiation monitor automatic actions.

SRO-Only Analysis

See attached SRO-only guidance flowchart. Internal EOP/AP procedure transition. Knowledge beyond simply entry conditions is required to arrive at the correct answer.

Answer Choice Analysis

A. INCORRECT. Both AP-24.00 and AP-24.01 clearly state that the correct transition is to AP-24.01 instead of ES-0.1. However, ES-0.1 is certainly a plausible choice, because once 1-E-0 is initiated, the RNO of step 4 directs a transition to ES-0.1, without any notes or cautions in the EOP about this particular case, where a transition to ES-0.1 is NOT desired.

B. INCORRECT. See analysis for A. above. Although AP-24.01 has specific steps to ensure the proper SJAE alignment, a note before step 1 of AP-24.01 specifically states that ES-0.1 must NOT be performed in parallel.

C. CORRECT. Even though 1-E-0 step 4 RNO directs a transition to 1-ES-0.1, the correct flow path is to transition from 1-E-0 to 1-AP-24.01. This is specified in AP-24.00, which has as step 2, "Initiate 1-E-0..." and as step 3, "GO TO 1-AP-24.01...." In 1-AP-24.01, step 13 RNO will realign the correct valves and ensure the automatic actions take place.

D. INCORRECT. Transitioning to 1-AP-24.01 is correct; however, one should not carry out AP-24.00 actions in parallel with AP-24.01. Step 3 of AP-24.00 specifies that if a Reactor trip is required, the operator must initiate 1-E-0 and GO TO 1-AP-24.01--that is, one is NOT to remain in AP-24.00. Once a reactor trip occurs and 1-AP-24.01 is entered, there is no other (re-)entry condition into AP-24.00.

NOTE: another possible wrong distractor could be "operators are required to be able to correct a radiation monitor automatic action failure from memory ("skill of the craft")" for the second part of choices "B" and "D;" see Lesson Plan ND-93.5-LP-1-DRR.

Supporting References

- 1-AP-24.00, "MINOR SG TUBE LEAK," rev 10, p. 2 and 3.

- 1-AP-24.01, "LARGE STEAM GENERATOR TUBE LEAK," rev 28, p. 2 and 7

- 1-E-0, "REACTOR TRIP OR SAFETY INJECTION," rev. 61, p. 3

-Surry lesson plan ND-93.5-LP-1, "PRE-TMI RADIATION MONITORING SYSTEM," rev 10, p. 2, 16, slide 7

References Provided to Applicant

none

Answer: C

4. 003AG2.4.31 8

Unit 1 Initial Conditions:

- Reactor power = 100%
- Control rod D-6 rod bottom light lit
- 1G-H2, RPI ROD BOTTOM $\leq$ 20 STEPS lit
- 0-AP-1.00 ROD CONTROL SYSTEM MALFUNCTION is entered

Based on the above conditions, which one of the following correctly states (1) if 0-AP-1.00 directs the initiation of 0-AP-23.00 RAPID LOAD REDUCTION to reduce power and (2) the parameter that is required to be monitored to reduce and stabilize power?

A. (1) Yes
(2) Loop ΔT

B. (1) Yes
(2) the highest reading PRNI

C. (1) No
(2) Loop ΔT

D. (1) No
(2) the highest reading PRNI

K/A

Dropped Control Rod: Knowledge of annunciator alarms, indications, or response procedures.

K/A Match Analysis

Requires knowledge of response procedures for a dropped control rod.

SRO-Only Analysis
Requires assessing plant conditions and then prescribing a procedure or section of a procedure to mitigate, recover, or with which to proceed. Knowledge above knowing entry conditions for APs is required.

Answer Choice Analysis

A. Incorrect; 1st part is incorrect because AP/1.00 does not reference AP/23 and AP/1.00 gives an hour to reduce power to 70-74%. 1st part is plausible because AP/23 is frequently used to reduce power during plant upsets. 2nd part is correct per a caution in AP/1.00 before step 17.

B. Incorrect; 1st part is incorrect because AP/1.00 does not reference AP/23 and AP/1.0
AP/23 is frequently used to reduce power during plant upsets.2nd part is incorrect because caution in AP/1.00 states that DT must be monitored during the ramp and used to stabilize power. 2nd part is plausible because the highest reading PRNI will
be more conservative than DT.

C. Correct: 1st part is AP/1.00 Step 17. A caution in AP/1.00 states that DT must be monitored during the ramp and used to stabilize power.

D. Incorrect; 1st part is correct. 2nd part is incorrect because caution in AP/1.00 states that DT must be monitored during the ramp and used to stabilize power. 2nd part is plausible because the highest reading PRNI will be more conservative than DT.

Supporting References
0-AP-1.00, ROD CONTROL SYSTEM MALFUNCTION

References Provided to Applicant
none

Licensee discuss the potential use of AP/23 for the power reduction.
Answer: C
5. 0054G2.2.25 1
Which one of the following correctly identifies two reasons for the Feedwater Line Isolation function, as specified in the bases of Technical Specification 3.7, "INSTRUMENTATION SYSTEMS?"

A. (1) Prevent excessive cooldown of the Reactor Coolant System; AND
(2) Reduces the consequences of a design basis steam generator tube rupture by preventing steam generator overfill.

B. (1) Prevent excessive moisture carry-over that could damage the main turbine blading; AND
(2) Reduces the consequences of a design-basis steam generator tube rupture by preventing steam generator overfill.

C. (1) Prevent excessive cooldown of the Reactor Coolant System; AND

(2) Reduces the consequences of a steam line break inside the containment by stopping the entry of main feedwater.

D. (1) Prevent excessive moisture carry-over that could damage the main turbine blading; AND
(2) Reduces the consequences of a steam line break inside the containment by stopping the entry of main feedwater.

K/A

Loss of Main Feedwater:
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
(CFR: 41.5 / 41.7 / 43.2) (SRO - 4.2)

K/A Match Analysis
The question is a straighforward link directly to the TS basis for feedwater isolation.

SRO-Only Analysis
See attached SRO-only flowchart. TS Basis knowledge required to arrive at correct answer.

Answer Choice Analysis

A. INCORRECT. The distractors are basically reasons for the HI-HI S/G level automatic function, worded to sound like the correct answers from the TS basis.

B. INCORRECT. see analysis of A. and C.

C. CORRECT. Answer is basically word-for-word from TS 3.7, which states: "The feedwater lines are isolated upon actuation of the SIS in order to prevent excessive cooldown of the Reactor Coolant System. This mitigates the effects of an accident such as a steam line break which in itself causes excessive temperature cooldown. Feedwater line isolation also reduces the consequences of a steam line break inside the containment by stopping the entry of feedwater."

D. INCORRECT. See analysis of A. and C.

Supporting References

-Surry Technical Specification 3.7, amendment nos. 180 and 180, p. 3.7-5 and 3.7-6

References Provided to Applicant

none

Answer: C
6. 0055G2.4.6 1
Unit 1 Initial Conditions:

- A steam generator tube rupture caused an automatic reactor trip and SI from 100% power.
- Operations personnel are performing actions in 1-E-3, "STEAM GENERATOR TUBE RUPTURE."

Current conditions:

- A maximum-rate cooldown using steam dumps to the condenser has begun.
- SI has just been reset.
- The RO reports that condenser vacuum is 28 " Hg and slowly lowering.
- The TSC informs the operations team that once all actions of E-3 are complete, it is required to implement the post-SGTR procedure that allows the FASTEST means of depressurizing the RCS and ruptured S/G.

Based on the current conditions, which one of the following is (1) a required action specified by E-3, AND (2) the correct post-SGTR procedure to implement?

A. (1) Ensure the condenser air ejector is aligned to containment, and then OPEN 1-SV-TV-102A.
(2) GO TO 1-ES-3.2, "POST-SGTR COOLDOWN USING BLOWDOWN."

B. (1) Ensure the condenser air ejector is aligned to containment, and then OPEN 1-SV-TV-102A.
(2) GO TO 1-ES-3.3, "POST-SGTR COOLDOWN USING STEAM DUMP."

C. (1) IF a Hi-CLS signal is NOT actuated, THEN realign the condenser air ejector for normal operations.
(2) GO TO 1-ES-3.2, "POST-SGTR COOLDOWN USING BLOWDOWN."

D. (1) IF a Hi-CLS signal is NOT actuated, THEN realign the condenser air ejector for normal operations.
(2) GO TO 1-ES-3.3, "POST-SGTR COOLDOWN USING STEAM DUMP."

K/A

055 Condenser Air Removal
Knowledge of EOP mitigation strategies. (as relating to the Condenser Air Removal system)
(CFR: 41.10 / 43.5 / 45.13) (SRO - 4.7)

K/A Match Analysis

The question requires the SRO applicant to demonstrate detailed knowledge of EOP mitigation strategies/transitions as related to expected effects of the condenser air removal system following an SI.

SRO-Only Analysis

See attached SRO-only flowchart.

Linked to SRO-only knowledge based on detailed internal EOP transition criteria and procedural selection outside of initial/entry conditions.

Answer Choice Analysis

A. INCORRECT. The lowering condenser vacuum is an expected condition. In the next few steps, 1-E-3 will ensure the proper operation of the air ejectors and mitigate the concern. Therefore the (1) part of this answer is correct. Part (2) is incorrect; the lesson plan for ES-3.3, "POST SGTR COOLDOWN USING STEAM DUMP," is very clear that it provides the fastest means of depressurizing the RCS and ruptured SG. ES-3.2 is plausible, if the applicant believes that the lowering condenser vacuum precludes the use of ES-3.3 through the steam dumps.

B. CORRECT. (1) Step 14 of 1-E-3 will align condenser air ejector to containment and improve the degraded vacuum condition. (2) is also correct; see analysis of A. above.

C. INCORRECT. (1) is incorrect, but plausible, because valve TV-SV-102 will (only) close automatically on a Hi-CLS signal. Also plausible because the question stem states that vacuum is lowering. Part (2) is also the incorrect procedural transition.

D. INCORRECT. (1) is incorrect choice, (2) is the correct proceural transition; see above analyses.

Supporting References

-Surry lesson plan ND-89.3-LP-2, "MAIN CONDENSATE SYSTEM," rev. 18, p. 11.

-1-E-3, "STEAM GENERATOR TUBE RUPTURE," rev. 38, p. 10, 12.

-Surry lesson plan ND-95.3-LP-16, "ES-3.3 POST SGTR COOLDOWN USING STEAM DUMP," rev. 12, p. 31.

References Provided to Applicant
none

Answer: B

7. 006A2.12 12

Initial plant conditions on Unit 1 are as follows:

- A SBLOCA has occurred.
- Radiation levels in the Auxiliary Building are increasing.
- The crew has transitioned to ECA-1.2 “LOCA Outside Containment”.
- The crew closed/verified closed SI-MOV-1890A and -1890B.
- RCS pressure was at 1700 psig and slowly dropping.

Current plant conditions on Unit 1 are as follows:

- The crew has closed SI-MOV-1890C.
- RCS pressure is at 1550 psig and slowly rising.

Which one of the following describes (1) the status of the LOCA and (2) the required procedure transition?

A. (1) LOCA has been isolated.
(2) Go to ECA-1.1, “Loss of Emergency Coolant Recirculation”.

B. (1) LOCA still exists.
(2) Go to ECA-1.1, “Loss of Emergency Coolant Recirculation”.

C. (1) LOCA has been isolated.
(2) Go to 1-E-1, “Loss of Reactor or Secondary Coolant”.

D. (1) LOCA still exists.
(2) Go to 1-E-1, “Loss of Reactor or Secondary Coolant”.

K/A

Emergency Core Cooling: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions requiring actuation of ECCS.

K/A Match Analysis

Requires applicant to predict the impact of a leak outside containment on the alignment of emergency core cooling equipment and perform the actions from ECA-1.2 for transitioning back to E-1.

SRO-Only Analysis

The question requires the applicant to assess plant conditions and know the intent of the specific steps to determine the correct procedural transition..

Answer Choice Analysis

A. In-Correct but plausible since the increasing RCS pressure indicates the leak has been isolated. In addition, the previous actions have closed all the cold and hot leg recirculation valves so it would seem plausible to transition to ECA-1.1, Loss of Emergency Coolant Recirculation". However, the correct action is to transition back to E-1.

B. In-Correct but plausible since the actions are correct if the leak still exists. However, the increasing RCS pressure indicates the leak has been isolated and the crew should transition to E-1.

C. Correct – The increasing RCS pressure indicates the leak has been isolated. The correct actions are to place LHSI pumps in PTL, close LHSI pump suction valves and transition to E-1.

D. In-Correct but plausible since reopening SI-MOV-1890C is correct if the leak still exists. The transition to E-1 is correct. However, the leak has been isolated.

Supporting References

ND-95.3-LP-21, "ECA-1.2, LOCA Outside Containment", Rev. 7, Obj. A

References Provided to Applicant

none

NOTE:Original question used on Surry 02-301 exam – developed by G. Laska (WE04G2.4.9). Modified conditions to indicate isolation of leak and asked for status of leak.

Answer: C

8. 0073A2.02 1

Unit 1 Initial Conditions:

- Holding at 30% power for chemistry, following a refueling outage.
- The Power Range NI input for 1-MS-RM-190, 1-MS-RM-191, and 1-MS-RM-192 (Main Steam Line N-16 radiation monitors) has failed to 100% power.

Current conditions:

- Annunciator 1A-A3, "N-16 HIGH," is NOT LIT
- Annunciator 1A-B3, "N-16 ALERT," is LIT
- Annunciator 1A-C3, "N-16 TROUBLE," is NOT LIT
- Annunciator 1D-E5, "CHG PP TO REGEN HX HI-LO FLOW," is LIT
- Pressurizer level is STABLE
- VCT level is STABLE

Based on the current conditions, which one of the following (1) is the correct procedural transition in accordance with the ARP for 1A-B3, "N-16 ALERT," AND (2) if no corrective actions have been taken for the power range NI input module, the alarm setpoints for 1-MS-RM-190 through -192 are _____. ?

A. (1) 0-OSP-RC-002, "STEAM GENERATOR PRIMARY TO SECONDARY LEAKAGE MONITORING."
(2) lower than normal.

B. (1) 1-AP-16.00, "EXCESSIVE RCS LEAKAGE."
(2) lower than normal.

C. (1) 1-AP-16.00, "EXCESSIVE RCS LEAKAGE."
(2) higher than normal.

D. (1) 0-OSP-RC-002, "STEAM GENERATOR PRIMARY TO SECONDARY LEAKAGE MONITORING."
(2) higher than normal.

K/A
Process Radiation Monitor (PRM) System
Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.
(CFR: 41.5/43.5/45.3/45.13) (SRO - 3.2)

K/A Match Analysis
Given a PRM detector failure condition, the SRO applicant will correctly determine the impact on the setpoints; and given an operationally valid situation, the SRO applicant will correctly apply/select procedures to correct, control, or mitigate the issue.

SRO-Only Analysis
This is an analysis level question since the candidate must analyze the impact of the power input to the detector circuitry failing high to determine the effect on the alarm setpoint.

This is an SRO only question linked to 10CFR55.43(b)(5). The question can NOT be answered using system knowledge alone. It can NOT be answered by knowing immediate actions, or basic procedure entry conditions (cover page material). To correctly answer this question, the candidate must assess plant conditions and then decide which procedure should be implemented.

Answer Choice Analysis

****

NOTE TO SURRY: Please validate the Power Range NI input part of this question with your actual plant response. The lesson plan for the N-16 monitors was not very detailed about power compensation.

****

A. INCORRECT. (1) The ARPs for both N-16 HIGH and N-16 ALERT specify to transition to 1-AP-16.00, "EXCESSIVE RCS LEAKAGE," on any of the following conditions: PRZR level - DECREASING; OR Annunciator 1D-E5, CHG PP TO REGEN HX HI-LO FLOW-LIT; OR A discernable negative change in VCT level trend has developed." 0-OSP-RC-002 is an incorrect, but plausible choice, because it would be correct if the annunciator 1D-E5 were NOT lit. (2) Due to much longer loop transport times at lower power, N-16 has more time to decay prior to reaching the area in the main steam lines adjacent to the monitors. Therefore, the alarm setpoint for a given leak must be lower than that for 100% power to ensure accuracy. Thus, (2) is incorrect for this distractor.

B. INCORRECT. (1) is correct choice, (2) incorrect. See above.

C. CORRECT. Both (1) and (2) correct as per the above.

D. INCORRECT. (1) is incorrect, (2) correct. See above analysis.

Supporting References

-modified from McGuire 2009-301 exam question SRO #94.

-Surry procedure 1A-A3, "N-16 HIGH," rev. 3.

-Surry procedure 1A-B3, "N-16 ALERT," rev. 3.

-Surry procedure 1A-C3, "N-16 TROUBLE," rev. 3.

References Provided to Applicant
none

Answer: C
9. 0076AA2.02 1
Unit 1 Initial Conditions:

- At time 0930, unexpected grid fluctuations caused an automatic turbine trip from 100% power.
- Chemistry personnel drew a post-trip RCS sample at time 1005.

- Control room operators have stabilized the unit at 547 °F and normal operating pressure.

Current conditions:

- At time 1045, a Chemistry supervisor reports that the post-trip RCS sample total specific activity reading is greater than the 100/(E bar) limit by 28%.

Based on the current conditions, which one of the following (1) is the correct time the LCO for Technical Specification (TS) 3.1.D, Maximum Reactor Coolant Activity, is NOT met; AND (2) the basis of the requirement to cool down the reactor to less than 500 °F, in accordance with TS 3.1.D?

A. (1) LCO not met at 1005;
(2) In the unlikely event of an assumed 30 minute radioactive release during the design-basis S/G tube rupture, the iodine partitioning factor below this RCS temperature ensures exposure limits are not exceeded at the site boundary.

B. (1) LCO not met at 1045;
(2) In the unlikely event of a design-basis S/G tube rupture, the saturation pressure corresponding to this RCS temperature is well below the pressure at which the atmospheric relief valves on the secondary side would be actuated.

C. (1) LCO not met at 1045;
(2) In the unlikely event of an assumed 30 minute radioactive release during the design-basis S/G tube rupture, the iodine partitioning factor below this RCS temperature ensures exposure limits are not exceeded at the site boundary.

D. (1) LCO not met at 1005;
(2) In the unlikely event of a design-basis S/G tube rupture, the saturation pressure corresponding to this RCS temperature is well below the pressure at which the atmospheric relief valves on the secondary side would be actuated.

K/A
High Reactor Coolant Activity
Ability to determine and interpret the following as they apply to High Reactor Coolant Activity: Corrective actions required for high fission product activity in RCS.
(CFR: 43.5/45.13) (SRO - 3.4)

K/A Match Analysis
The question requires the SRO applicant to correctly demonstrate knowledge of the Technical Specifications for RCS activity, as well as the basis for this specification.

SRO-Only Analysis
See attached SRO-only flow chart. TS Basis knowledge needed to arrive at correct

answer.

Answer Choice Analysis

A. INCORRECT. 1005 is the incorrect time, because the initial notification of the abnormality is considered the "start time" of inoperability. The second part of the answer is also incorrect; TS 3.1.D. basis states "Rupture of a steam generator tube would allow radionuclides in the reactor coolant to enter the secondary system. The limiting case involves a double-ended tube rupture coincident with loss of the condenser and release of steam from the secondary side to the atmosphere via the main steam safety valves or atmospheric relief valves. This is assumed to continue for 30 minutes in the analysis. The operator will take action to reduce the primary side temperature to a value below that corresponding to the relief or safety valve setpoint. Once this is accomplished the valves can be closed and the release terminated." However, the distractor is plausible, because everything associated with this specification is concerned with a release during a design basis tube rupture.

B. CORRECT. See above analysis. The statement about the saturation pressure and atmospheric relief valves is basically word-for-word from the TS.

C. INCORRECT. Incorrect time, wrong reason for RCS cooldown.

D. INCORRECT. See above analysis.

Supporting References

-SPS TS 3.1.D

References Provided to Applicant

Steam Tables

Answer: B

10. 010G2.4.20 12

Unit 1 initial conditions:

- Reactor power = 100%
- SGTR = 75 gpm on 1A SG
- Reactor is manually tripped
- 1C RCP trips

Current conditions:

- 1-E-3 (STEAM GENERATOR TUBE RUPTURE) is in progress
- It is determined that Pzr spray is not adequately reducing RCS pressure and the decision is made to use the PORV to reduce RCS pressure.

Based on the above conditions, which one of the following states: (1) the reason for minimizing the cycling of the PORV and (2) the procedure that 1-E-3 directs you to perform if the PORV and its associated block valve fail to close?

A. (1) To prevent rupturing the PRT
(2) 1-ECA 3.3 SGTR WITHOUT PRESSURIZER PRESSURE CONTROL

B. (1) To prevent rupturing the PRT
(2) 1-ECA 3.1 SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY

C. (1) To prevent the Tube rupture from degrading
(2) 1-ECA 3.3 SGTR WITHOUT PRESSURIZER PRESSURE CONTROL

D. (1) To prevent the Tube rupture from degrading
(2) 1-ECA 3.1 SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY

K/A
Pressurizer Pressure Control: Knowledge of the operational implications of EOP warnings, cautions, and notes.

K/A Match Analysis
Requires knowledge of EOP Cautions.

SRO-Only Analysis
Requires detailed knowledge of EOP steps having to do with securing PORV use when depressurizing the RCS.

Answer Choice Analysis

A Incorrect: 1st part is correct. 2nd part is plausible because it is criteria for closing the PORV if Pzr level is > 22%.

B Correct: The PORV relieves to the PRT so using the PORV will eventually cause the PRT rupture disk to rupture. Criteria for securing from using the PORV are:
Pzr level>69%
RCS subcooling < 30 ^{0}F
RCS press < Ruptured SG press AND Pzr level > 22%

C Incorrect: 1st part is plausible because the PORVs have failed to reseat (TMI) which constitutes a SBLOCA. 2nd part is plausible because it is criteria for closing the PORV if Pzr level is > 22%.

D Incorrect: 1st part is plausible because the PORVs have failed to reseat (TMI) which

constitutes a SBLOCA. 2nd part is correct.

Supporting References
1-E-3 Steam Generator Tube Rupture. ND-95.3-LP-13 Obj A & B

References Provided to Applicant
none

Answer: B
11. 015/17AG2.2.22 1
Initial plant conditions on Unit 2 are as follows:

- A power increase is in progress following reactor startup.
- Reactor power is at 8%.
- Pressurizer Spray valve PCV-455A cannot be opened.
- All three RCPs are operating.

Current plant conditions on Unit 2 are as follows:

- RCP 'C' trips on ground overcurrent.

Based on the above conditions, which one of the following describes whether action statements of the following LCOs are required to be performed:

- LCO 3.1.A.4, Reactor Coolant Loops
- LCO 3.1.A.5, Pressurizer

Action statement(s) of…
A. LCO 3.1.A.4 is/are required.
LCO 3.1.A.5 is/are NOT required.

B. LCO 3.1.A.4 is/are NOT required.
LCO 3.1.A.5 is required.

C. both LCO 3.1.A.4 and LCO 3.1.A.5 are required.

D. neither LCO 3.1.A.4 nor LCO 3.1.A.5 are required.

K/A
RCP Malfunctions

Knowledge of limiting conditions for operations and safety limits as it relates RCP Malfunctions.

K/A Match Analysis
Applicant must recognize that loss of RCP 'C' will require entry into both LCO 3.1.A.4. and 3.1.A.5.

SRO-Only Analysis
The question requires a knowledge of the T.S. bases associated with LCO 3.1.A.4 concerning what constitutes an in-service reactor coolant loop to determine whether actions from LCO 3.1.A.4 are required.

Answer Choice Analysis
A. In-Correct but plausible since LCO 3.1.A.4 would be entered given that LCO 3.1.A.4.b. states, *"POWER OPERATION with **less than three loops in service** is prohibited*.". However, LCO 3.1.A.5 would also be entered since LCO 3.1.A.5.a states, *"The reactor shall be maintained subcritical by at least 1% until the steam bubble is established and **the necessary sprays** and at least 125 KW of heaters are operable."* With PCV-455A inoperable, PCV-455B becomes inoperable once RCP 'C' trips.
B. In-Correct but plausible if the applicant believes that a running RCP is not required for an RCS loop to be considered in service. The second half of the answer is correct. LCO 3.1.A.5 would be entered since LCO 3.1.A.5.a states, *"The reactor shall be maintained subcritical by at least 1% until the steam bubble is established and **the necessary sprays** and at least 125 KW of heaters are operable."*
C. Correct –. Both LCO 3.1.A.4 and LCO 3.1.A.5 would be entered. *See previous distractor discussions for justification.*
D. In-Correct but plausible if the applicant believes that a running RCP is not required for an RCS loop to be considered in service AND does not recognized that both Pressurizer Spray valves are inoperable once RCP 'C' trips.

NOTE TO LICENSEE: The correct answer was based on discussions with facility SME. The Technical Specifications bases do not provide a specific discussion with regards to what constitutes a loop being in service per LCO 3.1.A.4. Please provide documentation as to what constitutes a loop being in service.
Also, neither LCO 3.1.A.5 nor its basis states that sprays have an impact on Technical Specifications once the reactor is above 1% subcritical. Please provide documentation for pressurizer operability when sprays are unavailable once a steam bubble is established and power is above 1% subcritical.

Supporting References
Technical Specification 3.1.A
Technical Specification 3.0.1
ND-88.1-LP-9, Technical Specifications Overview, Rev. 16, Obj. G

References Provided to Applicant

none

Answer: C

12. 025AA2.05 2

Unit 1 initial conditions:

Time = 0800
Plant was on RHR following shutdown for refueling
SGs are not available
RCS temperature = 190 ^{0}F stable
RHR flow = 2200 gpm
RCS level = 10 feet decreasing
AP/27 (LOSS OF DECAY HEAT REMOVAL CAPABILITY) has been initiated

Current plant conditions:

Time = 0825
1 CHG pump was started for RCS fill
RHR pumps have been secured
RCS level = 11.5 ft increasing
RCS temperature = 205 ^{0}F increasing

Based on the above conditions: (1) Classify the event using the Emergency Plan and (2) Once RHR is restored, state the maximum cooldown rate allowed per 1-AP-27?

(Reference Provided)

A. (1) Alert
(2) 25 ^{0}F/Hr

B. (1) Alert
(2) 50 ^{0}F/Hr

C. (1) Site Area Emergency
(2) 50 ^{0}F/Hr

D. (1) Site Area Emergency
(2) 25 ^{0}F/Hr

K/A

Loss of RHR: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Limitations on LPI flow and temperature rates of change.

K/A Match Analysis

~~Requires knowledge of limits on cooldown rate during loss of decay heat removal and~~

~~recovery~~.
Requires the ability to determine the emergency classification based on the reduction and eventual loss of RHR flow due to invetory loss and requires knowledge of plant cooldown limits once RHR is restored.

SRO-Only Analysis
Requires in depth knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Answer Choice Analysis

A. Incorrect: 1st part is incorrect because CS2 (Loss of Reactor Vessel inventory affecting core decay heat removal capability) existed = SAE. 1st part is plausible because CA 2 (Loss of RCS inventory) and CA3 (Inability to maintain plant in cold shutdown with irradiated fuel in the Reactor Vessel) apply. 2nd part is incorrect because 50 ^{0}F/Hr is the rate used for recovery once RHR is re-established. It is plausible because 25 ^{0}F/Hr is the cooldown rate for natural circulation cooldown in Attachment 4 of AP/27.

B. Incorrect: 1st part is incorrect because CS2 (Loss of Reactor Vessel inventory affecting core decay heat removal capability) existed = SAE. 1st part is plausible because CA 2 (Loss of RCS inventory) and CA3 (Inability to maintain plant in cold shutdown with irradiated fuel in the Reactor Vessel) apply. 2nd part is correct per 1AP/27 , Step 27.

C. Correct: 1st part is incorrect because CS2 (Loss of Reactor Vessel inventory affecting core decay heat removal capability) existed = SAE. 2nd part is correct per 1AP/27 , Step 27.

D. Incorrect: 1st part is incorrect because CS2 (Loss of Reactor Vessel inventory affecting core decay heat removal capability) existed = SAE. 2nd part is incorrect because 50 ^{0}F/Hr is the rate used for recovery once RHR is re-established. It is plausible because 25 ^{0}F/Hr is the cooldown rate for natural circulation cooldown in Attachment 4 of AP/27.

Supporting References
Surry Emergency Plan AP/27 (LOSS OF DECAY HEAT REMOVAL CAPABILITY)

References Provided to Applicant
Emergency Plan

Answer: C
13. 027AA2.15 4
Unit 1 initial conditions:
Time = 1000
Reactor power = 100%

PORV-1455C indicates open
Both Pzr Spray valves indicate open
RCS Pressure = 2200 psig decreasing
AP/31 (Increasing or Decreasing RCS Pressure) initiated

Current conditions:
Time = 1001
Reactor Power = 97%
RCS Pressure = 2100 psig increasing
Spray valve in MANUAL and closed
PORV- 1455C in MANUAL and closed

Based on the above conditions, which one of the following correctly states: (1) the component that failed high and (2) the status of PORV 1455C operability per Technical Specifications?

A. (1) P-444
(2) PORV is considered OPERABLE

B. (1) P-444
(2) PORV is NOT considered OPERABLE

C. (1) P-445
(2) PORV is considered OPERABLE

D. (1) P-445
(2) PORV is NOT considered OPERABLE

K/A
Pressurizer Pressure Control System Malfunction . Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: Actions to be taken if PZR pressure instrument fails high

K/A Match Analysis
Requires knowledge of how instrument failure affects the Pzr pressure control system and actions to mitigate the event.

SRO-Only Analysis
Requires ability to interpret plant conditions and select appropriate AP/EOP to mitigate the event.

Answer Choice Analysis
A. Incorrect. 1st part is correct. 2nd part is incorrect because the PORV is not able to
perform its Normal Function at power (prevent challenging the code safetys). 2nd

part is plausible because it is still operable in MANUAL.

B. Correct. Indications are indicative of transmitter P-444 failed high. TS directs the Block Valve for that PORV to be closed which renders the PORV inoperable. If the PORV was still operable, this action would not be required. In the TS Bases 3.1.5c, it states this action is taken when the PORV is Inoperable.

C. Incorrect 1st part is incorrect because this transmitter does not control all of the functions to create the parameters listed. It is plausible because P-445 controls a PORV and will cause RCS pressure to decrease. 2nd part is incorrect because the PORV is not able to perform its Normal Function at power (prevent challenging the code safetys). 2nd part is plausible because it is still operable in MANUAL.

D. Incorrect: 1st part is incorrect because this transmitter does not control all of the functions to create the parameters listed. It is plausible because P-445 controls a PORV and will cause RCS pressure to decrease. 2nd part is incorrect because the PORV is not able to perform its Normal Function at power (prevent challenging the code safetys). 2nd part is correct.

Supporting References
TS Section 3.1.4a
ND-93.3-LP5, Pzr Press Control pg 11 Obj: C

References Provided to Applicant
none

Licensee to determine operability of PORV
Answer: B
14. 035A2.01 17
Unit 1 initial conditions:

Reactor power = 100%
Main Steam Line Break inside containment occurs on the 1B SG
Maximum containment pressure reached = 4 psig
1-E-2 FAULTED STEAM GENERATOR ISOLATION is in progress

Current plant conditions:

RCS Pressure = 1750 psig increasing
RCS Subcooling = 95 ^{0}F increasing
A SG NR level = 15% increasing
B SG WR level = 5% stable
C SG NR level = 18% increasing
Pzr level = 35% increasing

(1) Which ONE of the following parts of the curve in TS Figure 3.8-1 is based on the peak calculated pressure criteria from this event and (2) based on the current plant

conditions, which procedure will 1E2 direct you to GO TO?

(Reference provided)

A. (1) Horizontal upper limit line (2) 1-ES-1.1 SI TERMINATION

B. (1) Horizontal upper limit line
(2) 1-E-1 LOSS OF REACTOR OR SECONDARY COOLANT

C. (1) Sloped line from 70-100 ^{0}F SW temp(2) 1-ES-1.1 SI TERMINATION

D. (1) Sloped line from 70-100 ^{0}F SW temp
(2) 1-E-1 LOSS OF REACTOR OR SECONDARY COOLANT

K/A
Steam Generator: Ability to (a) predict the impacts of the following malfunctions or operations on the S/GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulted or Ruptured S/Gs.

K/A Match Analysis
Requires knowledge of procedures used to mitigate a Faulted SG.

SRO-Only Analysis
Requires knowledge of Tech Spec bases that is required to analyze Tech Spec required actions and terminology.

Answer Choice Analysis

A. Correct: 1st part is correct per TS 3.8-4. 2nd part is correct per step 8 of 1E-2 FAULTED STEAM GENERATOR ISOLATION.

B. Incorrect: 1st part is correct per TS 3.8-4. 2nd part is incorrect because per 1E-2 Step 8 you meet the criteria to GO TO 1ES1 SI Termination. Plausible because if the Applicant thinks that Adverse Containment Conditions exist or if they did exist (> 5 psig), 1E-2 would direct you to GO TO 1E-1 LOSS OF REACTOR OR SECONDARY COOLANT.

C. Incorrect: 1st part is incorrect because it is based on LOCA depressurization criteria. 1st part is plausible because it is an upper limit on the curve. 2nd part is correct per step 8 of 1E-2 FAULTED STEAM GENERATOR ISOLATION..

D. Incorrect: 1st part is incorrect because it is based on LOCA depressurization criteria. 1st part is plausible because it is an upper limit on the curve. 2nd part is incorrect because per 1E-2 Step 8 you meet the criteria to GO TO 1ES1 SI Termination. Plausible because if the Applicant thinks that Adverse Containment Conditions exist or if they did exist (> 5 psig), 1E-2 would direct you to GO TO

1E-1 LOSS OF REACTOR OR SECONDARY COOLANT.

Supporting References
1-E-2
ND-95.3-LP-12, E-2 Obj: A
TS 3.8 Containment
ND-95.3-LP-3 E-0, pg 8 Adverse Containment Criteria

References Provided to Applicant
TS Figure 3.8-1

Answer: B
15. 051G2.4.11 9
Unit 1 initial conditions:
Time = 1500
Reactor power = 100 %
A loud explosion is heard from the main turbine area (Security reports that no suspicious activity noted)
Condenser Vacuum = 27" Hg decreasing
1AP/14 (LOSS OF MAIN CONDENSER VACUUM) initiated

Current plant conditions:
Time = 1510
Reactor Power = 60%
Condenser vacuum = 25" Hg decreasing
An operator reports that there was insulation on fire around a Reheat Stop valve. The fire is out but he hears a hissing noise

Based on current plant conditions, which one of the following correctly states: (1) the procedure that will be used to continue the load reduction and (2) the e-plan classification?

(Reference provided)

A. (1) 1AP/14 Attachment 2 RAMPING AT GREATER THAN OR EQUAL TO 1%/MIN
(2) UNUSUAL EVENT

B. (1) 1AP/14 Attachment 2 RAMPING AT GREATER THAN OR EQUAL TO 1%/MIN
(2) ALERT

C. (1) 1AP/23 RAPID LOAD REDUCTION
(2) UNUSUAL EVENT

D. (1) 1AP/23 RAPID LOAD REDUCTION
(2) ALERT

K/A
Loss of Condenser Vacuum: Knowledge of abnormal condition procedures.

K/A Match Analysis
Requires knowledge of abnormal procedures.

SRO-Only Analysis
Requires ability to assess plant conditins and then prescribing a procedure or section of a procedure to mitigate, recover or with which to proceed.

Answer Choice Analysis

A. Incorrect: 1st part incorrect because in attachment 2 of AP14 it states that if power decreases to 60% and further power reduction is anticipated, THEN initiate AP/23.
1st part is plausible because AP/14 Attachment 2 is used for the power reduction to this point. 2nd part is correct based on Fire/Explosion in the protected area boundary.

B. Incorrect: 1st part incorrect because in attachment 2 of AP14 it states that if power decreases to 60% and further power reduction is anticipated, THEN initiate AP/23.
1st part is plausible because AP/14 Attachment 2 is used for the power reduction to this point. 2nd part is plausible because it is a fire affecting a normal shutdown (the condenser) but incorrect because the condenser is not required to establish or maintain safe shutdown.

C. Correct: 1st part is correct in that in attachment 2 of AP14 it states that if power decreases to 60% and further power reduction is anticipated, THEN initiate AP/23.
2nd part is correct based on Fire/Explosion in the protected area boundary.

D. Incorrect: 1st part is correct in that in attachment 2 of AP14 it states that if power decreases to 60% and further power reduction is anticipated, THEN initiate AP/23.
2nd part is plausible because it is a fire affecting a normal shutdown (the condenser) but incorrect because the condenser is not required to establish or maintain safe shutdown.

Supporting References
AP/14 LOSS OF MAIN CONDENSER VACUUM.
Emergency Plan
ND-95.1-LP-6 Obj: B

References Provided to Applicant
Emergency Plan SEP

Licensee to determine how much of SEP to be provided.
Answer: C

16. 059G2.4.14 14
Unit 1 initial conditions:
Reactor power = 100%
Loss of offsite power occurs
Reactor trip
Both EDGs start but both output breakers fail to close
TD AFW pump fails to start
1-ECA-0.0 LOSS OF ALL AC POWER has been initiated

Based on the above conditions, which one of the following correctly states (1) the EOP that will direct supplying AFW to the SG's and (2) whether the initial conditions coincide with the conditions for the loss of auxiliary feedwater design basis accident as stated in Tech Spec Bases 3.6, TURBINE CYCLE?

A. (1) 1-ECA-0.0 before directing emergency buses to be energized
(2) No

B. (1) 1-FR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK after emergency busses are energuzed
(2) No

C. (1) 1-ECA-0.0 before directing emergency buses to be energized
(2) Yes

D. (1) 1-FR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK after emergency busses are energuzed
(2) Yes

K/A
Main Feedwater: Knowledge of general guidelines for EOP usage.

K/A Match Analysis
Requires knowledge of how the EOP directs feedwater restoration after a loss of all feedwater.

SRO-Only Analysis
Requires detailed knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures. This beyond knowing CSF path selection.

Answer Choice Analysis
A. Correct: 1-ECA-0.0 will direct getting AFW flow to the SGs after verifying Rx and Turbine trip. TS design bases accident for AFW is a loss of Main Feedwter with On
Site power (RCP's running)

B. Incorrect: 1st part is incorrect because ECA-0.0 is a higher priority section of the EOP and it directs restoration fo AFW. 1st part is plausible because it will address the loss of feedwater after ECA-0.0 is exited. 2nd part is correct.

C. Incorrect: 1st part is correct. 2nd part is incorrect because TS design bases accidtne for AFW is a loss of Main Feedwater with On site Power (RCPs running). Plausible because you do not have the TD AFW pump.

D. Incorrect: 1st part is incorrect because ECA-0.0 is a higher priority section of the EOP and it directs restoration fo AFW. 1st part is plausible because it will address the loss of feedwater after ECA-0.0 is exited. 2nd part is incorrect because TS design bases accidtne for AFW is a loss of Main Feedwater with On site Power (RCPs running). Plausible because you do not have the TD AFW pump.

Supporting References
Ref:
1-FR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK
1-ECA-0.0 LOSS OF ALL AC POWER
TS 3.6

References Provided to Applicant
none

Answer: A
17. 062AA2.06 1
Initial plant conditions:
Unit 2 shutdown with fuel offloaded
Unit 1 = 100% power

Current plant conditions:
Annunciator 1D-G5, SW OR CC PPS DISCH TO CHRG PPS LO PRESS is in
alarm
1AP/12 SERVICE WATER SYSTEM ABNORMAL CONDITIONS has been initiated
Unit 1 operating CHG pump bearing temperatures:
1420 = 170 ^{0}F
1430 = 175 ^{0}F
1440 = 180 ^{0}F
1450 = 185 ^{0}F
1500 = 190 ^{0}F

Based on the above conditions: (1) which one of the following states the time at which 1-AP-12 directs shifting the operating charging pump and, (2) if all Unit 1 charging pumps are lost, correctly state the Tech Spec bases for using the designated Unit 2 charging pump?

A. (1) 1440
(2) To bring the operating unit to cold shutdown

B. (1) 1440
(2) To bring the operating unit to hot shutdown

C. (1) 1450
(2) To bring the operating unit to cold shutdown

D. (1) 1450
(2) To bring the operating unit to hot shutdown

K/A
Loss of Nuclear Svc Water:
The length of time after the loss of SWS flow to a component before that component may be damaged.

K/A Match Analysis
Requires knowledge of temperature limits on components supplied by SWS.

SRO-Only Analysis
Requires knowledge of Tech Spec bases that is required to analyze Tech Spec required actions and terminology.

Answer Choice Analysis

A. Correct: At 180 ^{0}F, AP/12 directs the charging pumps to be shifted. Per TS 3.2 C&VCS for a shutdown unit, one charging pump with a source of borated water shall be available for cross-connect with the operating unit so that if the operating units charging pumps become inoperable, the shutdown units charging pump can bring the disabled unit to cold shutdown.

B. Incorrect: 1st part is correct because at 180 ^{0}F, AP/12 directs the charging pumps to be shifted. 2nd part is not correct because TS 3.2 states the shutdown units charging pump is used to bring the diabled unit to cold shutdown. 2nd part is plausible because being in hot shutdown would put the plant in a stable condition while repairs are conducted.

C. Incorrect: 1st part is incorrect because per AP/12 directs them to be shifted at 180 ^{0}F. 1st part is plausible because at 185 ^{0}F, AP/12 directs the charging pump to be secured. 2nd part is correct.

D. Incorrect: 1st part is incorrect because per AP/12 directs them to be shifted at 180 ^{0}F. 1st part is plausible because at 185 ^{0}F, AP/12 directs the charging pump to be secured. 2nd part is not correct because TS 3.2 states the shutdown units charging pump is used to bring the diabled unit to cold shutdown. 2nd part is

plausible because being in hot shutdown would put the plant in a stable condition while repairs are conducted.

Supporting References
TS 3.2, AP/12 Step 4 & 5, ND-89.5-LP-2 Obj H

References Provided to Applicant
none

Answer: C
18. 079G2.2.22 1
Given the following plant conditions:

- Unit 1 is at 100%
- A loss of Containment Instrument Air has occurred
- 1B-F6, CTMT INST AIR HDR LO PRESSURE, annunciates
- 1D-C6, PRZR PWR RELIEF VV LO AIR PRESS, annunciates
- Containment Instrument Air was crosstied with Instrument Air
- Containment Instrument Air Pressure = 85 psig and increasing
- All PORV air bottles are properly aligned with air pressures of 1050 psig

Which one of the following correctly states (1) the status of LCO 3.1.A.6, "PORV Operability" and (2) the Tech Spec required operator actions, if any?

A. (1) The LCO is met.
(2) No further action associated with the PORVs is required.

B. (1) The LCO is met.
(2) Verify PORV operability by closing PORV Block Valves, manually cycle the PORVs, and then re-open the PORV Block Valves.

C. (1) The LCO is NOT met.
(2) Restore the PORV backup air supply within 14 days OR be in HSD within the next 6 hours.

D. (1) The LCO is NOT met.
(2) Close and remove power from both PORV block valves within one hour AND be in HSD within the next 6 hours

079 Station Air
G2.2.22: Knowledge of limiting conditions for operations and safety limits

K/A MATCH ANALYSIS:
The question requires knowledge of PORV operability which is impacted by a loss of air. The operability determination causes the conditions of the LCO to not be met.

SRO-ONLY ANALYSIS:
Operability is primarily an SRO function unless the determinatation is made at a very basic level (I.E. if a pump is broke, it is obviously inop - which would be RO knowledge). This question requires the SRO to understand how the loss of instrument air affects the PORV operability, even when the PORV is available for use with cross-tied air.

Answer Choice Analysis:
A. Incorrect per 1D-C6 CTMT Inst Air P must be > 80 psig for the PORVs to be operable.
B. Incorrect because (per 1D-C6) with CTMT Inst Air P < 80 psig, the PORVs are inoperable.
C. Correct because PORVs are capable of being manually cycled with CTMT Inst Air P > 80 psig. The PORVs are INOP due to INOP air supply and you start a 14 day LCO clock.
D. Incorrect, the PORV is INOP but can be manually cycled. This choice is correct if the PORV could NOT be manually cycled. This would be a 1 hr LCO.

Surry Requal Bank Question #571 (LARP0001) & 2004-301 NRC Exam

References:
ND-92.1-LP-1, Station Air Systems, Rev. 13
ND-88.1-LP-3, Pressurizer and Pressure Relief, Rev. 12
1B-F6, CTMT INST AIR HDR LO PRESS, Rev. 1
1D-C6, PRZR PWR RELIEF VV LO AIR PRESS, Rev. 4
Technical Specification 3.1.A.6.c, Reactor Coolant System / Relief Valves

Answer: C
19. G2.1.20 13
Initial plant conditions on Unit 2 are as follows:

- Reactor power is 100%.
- A 20 gpd leak exists on steam generator ‘B’.

Current plant conditions on Unit 2 are as follows:

- Charging flow has slowly increased.

Auto-makeup to VCT has started.

VCT level is 29% and slowly rising.

Pressurizer level is stable at 54%.

Pressurizer pressure is stable at 2225 psig.

Crew has entered AP-16, “Excessive RCS Leakage”.

Radiation levels on MSL “B” show a slow increasing trend.

- The leak rate has been calculated at 12 gpm. [MAY NEED TO RAISE LR - DISCUSS WITH LICENSEE]

[REVIEW ALL THE CONDITIONS IN THE STEM WITH THE LICENSEE]

Which one of the following describes (1) whether the following procedure transition is required AND (2) the correct classification for the event?

Transition to 2-AP-24.00, “Minor SG Tube Leak” is…

(Reference provided)

A. (1) required.
(2) Alert

B. (1) NOT required.
(2) Alert

C. (1) required.
(2) NOUE

D. (1) NOT required.
(2) NOUE

[DISCUSS WITH THE LICENSEE TO DETERMINE CONDITIONS FOR THE STEM THAT WILL ENSURE ONE AND ONLY ONE CORRECT ANSWER AS WELL AS PLAUSIBILITY FOR THE DISTRACTORS]

K/A
Generics: Ability to interpret and execute procedure steps.

K/A Match Analysis
Requires applicant to interpret the leak indications, determine if transition to 1-AP-24.00 is required and determine the correct emergency classification associated with the leak.

SRO-Only Analysis
The question requires the applicant to correctly determine if a procedure transition is required from AP-16-00 and classify the event per the emergency plan. Both of which would require SRO- Only knowledge to determine.

Answer Choice Analysis
A. In-Correct but plausible since a procedure transition to 1-AP-24.00 is required.

B. In-Correct but plausible since a procedure transition would not be required if the applicant didn't recognize that MSL 'B' radition were increasing.

C. Correct – Transition to 1-AP-24.00 is required.

D. In-Correct. See above.

Supporting References
1-AP-16.00, Excessive RCS Leakage, Rev. 16
Emergency Plan, Rev. 54

References Provided to Applicant
Emergency Plan

NOTE: Facility reviewers please validate that the correct emergency classification was determined.
Answer: C
20. G2.2.14 20
Plant conditions:
RCS cooldown in progress
RCS temperature = 350 ºF decreasing
RCS pressure = 300 psig

Based on the above conditions in regards to the Overpressure Mitigation System (OMS),
(1) which one of the following correctly describes the required equipment configuration for the 72 hours following RCS temperature decreasing below 350 ºF and (2) what is the TS basis for that configuration? (Consider No TS modifications, LCOs...)?

A. (1) Pzr level is limited to 33%
(2) This is to allow the operator 10 minutes to take action from inadvertent initiation of full (3 pump) charging flow.

B. (1) Two PORVs are required to remain operable
(2) This is based on the PORVs ability to relieve RCS pressure from the start of a RCP with SG temp > RCS temp.

C. (1) Accumulators must be depressurized to less than the PORV setpoint
(2) This is to prevent exceeding the PORV capability if an inadvertent OMS initiation occurs.

D. (1) All but one charging pump shall be removed from service and incapable of injecting into the RCS
(2) This is to ensure any mass addition can be relieved by one PORV.

K/A
Knowledge of the process for controlling equipment configuration or status.

K/A Match Analysis
Requires knowledge of the equipment configuration for specific plant conditions.

SRO-Only Analysis
Requires knowledge of the plant configuration for cooldown operations and the TS Bases for that configuration.

Answer Choice Analysis
A. Incorrect: Plausible because the limit is correct but based on only one charging pump injecting.
B. Incorrect: 2 PORVs are required for the 1st 72 hours if no vent exists or Pzr level < 33%. Plausible because the bases stated is for one PORV being operable.
C. Incorrect: Accumulators can be isolated and valves de-energized as an alternative to depressurizing. While initiation may cause the PORV to lift, it will not exceed its capacity. Plausible because depressurizing the accumulators is an option to isolating them.
D. Correct. Per TS 3.1.G

Supporting References
ND-93.3-LP-6 Obj: E
TS3.1.G

References Provided to Applicant
none

Answer: D
21. G2.2.22 1
Which one of the following describes how the potential reactivity effects due to Reactor Coolant System cooldown during and following loop backfill are limited to acceptable levels, as specified in the Bases to Technical Specification 3.17, "LOOP STOP VALVE OPERATION?"

A. (1) There is a small absolute value of the isothermal temperature coefficient of reactivity at cold and refueling shutdown conditions.

(2) Reactivity effects due to boron stratification in the backfilled loop are NOT a concern, because stratification is NOT expected to take place at the normal shutdown boron concentrations and temperatures during the time to complete backfill of the loop and open the loop stop valves fully.

B. (1) There is a large absolute value of the fuel temperature coefficient of reactivity at cold and refueling shutdown conditions.

(2) Reactivity effects due to localized boron stratification in the backfilled loop are a concern; the requirements on relief line flow and boron concentration of the reactor coolant pump seal injection source are designed to mitigate any adverse effects of localized boron stratification.

C. (1) There is a small absolute value of the isothermal temperature coefficient of reactivity at cold and refueling shutdown conditions.

(2) Reactivity effects due to localized boron stratification in the backfilled loop are a concern; the requirements on relief line flow and boron concentration of the reactor coolant pump seal injection source are designed to mitigate any adverse effects of localized boron stratification.

D. (1) There is a large absolute value of the fuel temperature coefficient of reactivity at cold and refueling shutdown conditions.

(2) Reactivity effects due to boron stratification in the backfilled loop are NOT a concern, because stratification is NOT expected to take place at the normal shutdown boron concentrations and temperatures during the time to complete backfill of the loop and open the loop stop valves fully.

K/A
Knowledge of limiting conditions for operations and safety limits.
(CFR: 41.5/43.2/45.2) (SRO - 4.7)

K/A Match Analysis
The K/A is a Tier 3, or "generic" K/A. The question asks the SRO candidate to demonstrate knowledge of the bases for an important Technical Specifications LCO for Loop Stop Valve Operation.

SRO-Only Analysis
-see attached flowchart from SRO-only guidance document. TS Basis knowledge needed to arrive at the correct answer.

Answer Choice Analysis

A. CORRECT. Both choices (1) and (2) are taken word-for-word from the bases of TS 3.17, "LOOP STOP VALVE OPERATION," p. TS 3.17-7.

B. INCORRECT. (1) is plausible because it uses the exact same language of the correct version of (1), but is incorrect because a large negative value of the Doppler coefficient would be worse from a reactivity standpoint when considering cold shutdown/refueling conditions. (2) is also incorrect, but plausible, because it specifies that only localized boron stratification is a concern, and also because it mentions (correctly) limits placed on relief line flow rates and time, as well as limits placed on

boron concentration of the reactor coolant pump seal injection source, which are actually contained in the TS 3.17.

C. INCORRECT. (1) is correct version; (2) is the incorrect distractor.

D. INCORRECT. (1) is incorrect distractor; (2) is correct version.

Supporting References

SPS TS 3.17 and bases, especially p. 7.

References Provided to Applicant

None

Answer: A

22. G2.3.12 1

Unit 1 initial conditions:

Date = 6/24
Time = 0800
Reactor power = 100%
Waste gas storage tank activity level is reported which exceeds TS 3.11, Radioactive Gas Storage, limits

Current conditions:

Date = 6/26
Time = 0800
Reactor power = 100%
Waste gas storage tank activity level still exceeds Tech Spec 3.11 limits

Based on the above conditions, which one of the following correctly states: (1) if Tech Spec 3.0.1 is applicable and (2) the whole body dose that the tank radioactivity limit is designed to prevent exceeding at the exclusion area boundary if the tank were released IAW Tech Spec Basis?

A. (1) Yes
(2) 50 mrem

B. (1) Yes
(2) 0.5 rem

C. (1) No
(2) 50 mrem

D. (1) No
(2) 0.5 rem

K/A
Knowledge of radiological safety principles pertaining to licensed operator | duties, such as containment entry requirements, fuel handling responsibilities, | access to locked high-radiation areas, aligning filters, etc.

K/A Match Analysis
Requires knowledge of radiological limits associated with the health and safety of the public and how to apply technical specifications to stay within those limits.

SRO-Only Analysis
Requires knowledge of the facility operation limitations in the technical specifications and their bases.

Answer Choice Analysis
A. Incorrect: In TS 3.11.B.3 it states that the requirements fo Specification 3.0.1 are not applicable. 1st part is plausible because the time for condition 3.11.B.2 has expired. 2nd part is incorrect because in the TS bases 3.11 it states 0.5 rem. 2nd part is plausible becasue the Surry adminestrative limit for site visitors is 50 mrem.
B. Incorrect: In TS 3.11.B.3 it states that the requirements fo Specification 3.0.1 are not applicable. 1st part is plausible because the time for condition 3.11.B.2 has expired. 2nd part is correct per TS 3.11 bases.
C. Incorrect: 1st part is correct. 2nd part is incorrect because in the TS bases 3.11 it states 0.5 rem. 2nd part is plausible becasue the Surry adminestrative limit for site visitors is 50 mrem.
D. Correct: In TS 3.11.B.3 it states that the requirements fo Specification 3.0.1 are not applicable. In the tech spec bases for TS 3.11 it states it limited to the quantity which provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem in an event.

Supporting References
TS 3.11
ND-81.2-LP3

References Provided to Applicant
none

Answer: D
23. G2.3.4 22
Unit 1 initial plant conditions:
Reactor power = 50%
Plant shutdown in progress due to RCS activity greater than TS limits

AFW Pump 1FW 3B OOS

Current plant conditions:

'A' SG tube rupture occurs

'A' SG pressure = 1000 psig

Reactor has been tripped

1-E-3 STEAM GENERATOR TUBE RUPTURE in progress

The TSC has been established

An operator is dispatched to close 1-MS-87 (steam from the A SG to the TD AFW pump) in order to save valuable equipment

Based on the above conditions, which one of the following: (1) states the allowable dose (TEDE) the operator can receive while isolating steam to the TD AFW pump and (2) if the valve can not be closed, what procedural actions shall be taken IAW 1-E-3 to mitigate the failure?

A. (1) 10 Rem
(2) Remain in 1-E-3 and trip the TD AFW pump overspeed trip valve.

B. (1) 10 Rem
(2) GO TO 1-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY.

C. (1) 5 Rem
(2) Remain in 1-E-3 and trip the TD AFW pump overspeed trip valve.

D. (1) 5 Rem
(2) GO TO 1-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY.

K/A
Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Match Analysis
Requires knowledge of exposure limits under emergency conditions.

SRO-Only Analysis
Requires knowledge of EOP procedures and transition points.

Answer Choice Analysis

A. Correct: Allowable dose for equipment = 10 Rem. Per 1-E-3, if at least 1 motor driven AFW pump available, trip the TD AFW pump.

B. Incorrect: 1st part is correct. 2nd part is plausible because if the SG with the rupture could not be isolated from both of the intact SGs, it would be correct.

C. Incorrect: 1st part is plausible because the exposure could be counted towards a

PSE (the PSE limit is 5 Rem / yr). 2nd part is correct.

D. Incorrect: 1st part is plausible because the exposure could be counted towards a PSE (the PSE limit is 5 Rem / yr). 2nd part is plausible because if the SG with the rupture could not be isolated from both of the intact SGs, it would be correct.

Supporting References
ND-81.2-LP-3 Obj: E
1-E-3
ND-95.3-LP-13 E-3 Obj: A

References Provided to Applicant
none

Answer: A
24. G2.4.30 1
Unit 1 Initial Conditions:

- Holding at 30% power for fuel conditioning following a refueling outage.

Current conditions:

- Technicians performing a routine surveillance test on the AMSAC logic system indavertantly cause a half-train Train "A" AMSAC signal to be generated.
- Annunciator F-B-3, AMSAC INITIATED, is lit
- The technicians are able to reset the Train "A" AMSAC signal in ten (10) seconds.

Based on the current conditions, which one of the following correctly describes (1) whether the half-train AMSAC signal should be considered a VALID or INVALID actuation, as defined by VPAP-2802, "Notifications and Reports," AND (2) the most restrictive time requirement to report this event to the NRC, as specified by VPAP-2802?

(Reference provided)

A. (1) VALID actuation
(2) 4 hour notification

B. (1) INVALID actuation
(2) 8 hour notification

C. (1) VALID actuation
(2) 8 hour notification

D. (1) INVALID actuation
(2) 4 hour notification

K/A
Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.
(CFR: 41.10/43.5/45.11) (SRO - 4.1)

K/A Match Analysis
The question requires the applicant to demonstrate knowledge of the definitions inherent in the notifications procedure ("system operation/status"), and also show an ability to use the procedure to determine the correct time requirements for the given plant conditions, which are operationally valid.

SRO-Only Analysis
This question requires the applicant to know the definitions inherent in the Notifications procedure, and to apply them in a practical setting. Therefore, it is a higher-level comprehension/analysis question that is linked to 10CFR55.43(b)(1), "conditions and limitations in the facility license," in that ROs are not required to know and be able to apply reporting requirements.

Answer Choice Analysis

A. INCORRECT. (1) Surry/Dominion procedure VPAP-2802, "Notifications and Reports," section 4.3 specifies that a VALID actuation must result "from an intentional manual initiation or from a signal that was initiated in response to actual plant conditions or parameters satisfying the requirements for initiation, unless part of a preplanned test." For the given conditions, the inadvertant AMSAC actuation was caused as a result of testing, the actuation was a result of human error and was not pre-planned to occur, and was not in response to actual plant conditions. Therefore, to state that the actuation was VALID is plausible. (2) 4 hours is the most restrictive notification, to report an RPS actuation on a critical reactor.

B. INCORRECT. (1) VPAP-2802 section 4.2 specifies that an invalid actuation "is one that does not meet the criteria for being valid and are initiated for reasons other than to mitigate the consequences of an event (e.g., as part of a planned evolution, with the system properly removed from service, or after the safety function has already been completed). Invalid actuations include circumstances where instrument drift, spurious signals, human error, or other invalid signals caused actuation (e.g. jarring a cabinet, an error in the use of jumpers or lifted leads, an error in the actuation of switches or controls, equipment failure, radio frequency interference)." For the given conditions, human error caused the actuation; therefore INVALID actuation is correct. The candidate must then infer from the question whether the reactor tripped (yes). (2) Based on the provided reference material, the candidate my incorrectly choose an 8-hour notification based on auxiliary feedwater auto-start, if he/she incorrectly believes that the AMSAC actuation at a low power level would not produce a reactor trip (or only

trip the turbine and not the reactor as well). The plausibility of this choice is enhanced by the question stem stating that the signal is reset within 10 seconds (where a normal AMSAC signal is required to remain "in" for 27 seconds to cause an actuation).

C. INCORRECT. "VALID" actuation is wrong as per the above.

D. CORRECT. "INVALID" actuation is correct as per the above. VPAP-2802 section 6.3.4.a.3. states that a 4-hour report is required for "Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when actuation results from and is part of a pre-planned sequence during testing or reactor operation." In this case, an automatic reactor trip/RPS actuation did occur with the reactor critical. The reactor trip was not pre-planned; rather, it was caused by human error, and therefore the exclusion clause does not apply.

Supporting References

-VPAP-2802, "Notifications and Reports," rev 30, (p. 20, p. 82, and p. 86)

- Surry lesson plan ND-93.3-LP-17, "ANTICIPATORY MITIGATING SYSTEM ACTUATING CIRCUITRY (AMSAC)," rev. 11, p. 7 and 9.

References Provided to Applicant

-VPAP-2802, "Notifications and Reports," pages 79-91.

Answer: D

25. G2.4.9 24

Unit 1 plant conditions:

Time = 0200
RCS cooldown in progress
RCS temperature = 250 ^{o}F
RCS pressure = 320 psig
1A charging pump is the only running charging pump

Current plant conditions:

Time = 0210
RCS pressure = 280 psig decreasing
The maximum charging flow achieved with the 1A charging pump is 125 gpm

Based on the above conditions, which ONE of the following: (1) states the correct procedure to be entered and (2) what actions are directed by that procedure?

A. (1) 1AP-16.00 EXCESSIVE RCS LEAKAGE

(2) Align charging pump suction to the RWST

B. (1) 1AP-16.00 EXCESSIVE RCS LEAKAGE
(2) Align and start 1B and 1C charging pumps

C. (1) 1-AP-16.01 SHUTDOWN LOCA
(2) Align charging pump suction to the RWST

D. (1) 1-AP-16.01 SHUTDOWN LOCA
(2) Align and start 1B and 1C charging pumps

K/A
Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

K/A Match Analysis
Requires knowledge of shutdown procedures/mitigation strategies during an accident.

SRO-Only Analysis
Requires in depth knowledge of abnormal procedure guidelines and selection based on plant conditions.

Answer Choice Analysis
A. Incorrect: Note at the top of AP/16.00 states "If SI Accumulators are isolated, 1-AP-16.01, SHUTDOWN LOCA, should be used for guidance". Plausible because if > 350 ^{0}F, it would be correct. 2nd part is correct.
B. Incorrect: Note at the top of AP/16.00 states "If SI Accumulators are isolated, 1-AP-16.01, SHUTDOWN LOCA, should be used for guidance". 2nd part is plausible because if > 350 ^{0}F charging pumps would used as necessary per AP/16.00 (OPMG not in service). Having OPMG in service requires only 1 Chg available to inject into the RCS.
C. Correct. If SI Accumulators are isolated, 1-AP-16.01, SHUTDOWN LOCA, should be used for guidance. Being < 350 ^{o}F requires the accumulators to be isolated. 2nd part is step 8 d RNO.
D. Incorrect: 1st part is correct. 2nd part is plausible because if > 350 ^{0}F charging pumps would used as necessary per AP/16.00 (OPMG not in service). Having OPMG in service requires only 1 Chg available to inject into the RCS.

Supporting References
Ref: AP/16.00, AP/16.01

References Provided to Applicant
none

Answer: C