76. Given the following plant conditions:
    - The Plant is in Mode 6 for refueling
    - The Refueling team is making preperations to remove the first fuel assembly from the core
    - Refueling Cavity Level is at 23' 6"
    - Both 'A' and 'B' RHR pumps are in operation and aligned for Shutdown Cooling,
    - 'B' RHR pump has now tripped on overcurrent

    In accordance with Technical Specifications, the MINIMUM RHR flowrate for the above conditions is __(1)__ gpm AND the basis for the flow requirement is that this circulation flow rate is maintained through the core to __(2)__

    A. (1) 900 gpm

       (2) minimize the effect of a boron dilution incident and prevent boron stratification.

    B. (1) 900 gpm

       (2) reduce the possibility of cavitation during operation of the RHR pumps and ensures sufficient mixing in the event of a MODE 6 boron dilution incident.

    C. (1) 2500 gpm

       (2) minimize the effect of a boron dilution incident and prevent boron stratification.

    D. (1) 2500 gpm

       (2) reduce the possibility of cavitation during operation of the RHR pumps and ensures sufficient mixing in the event of a MODE 6 boron dilution incident.

Plausibility and Answer Analysis

Reason answer is correct: *Technical Specification 3.9.8.1 requires at least one RHR loop to be operable and in operation while in MODE 6 with fuel in the vessel when the water level above the top of the Rx vessel flange is ≥ 23'. The TS basis for this flow rate is that it is sufficient coolant circulation to minimize the effect of a boron dilution incident and prevent boron stratification.*

A. Incorrect. The first part is plausible since this is the flow rate that would be required IF the vessel water level were < 23' above the flange. The second part is the correct Tech Spec basis.

B. Incorrect. The first part is plausible since this is the flow rate that would be required IF the vessel water level were < 23' above the flange. The second part is plausible since this is the basis for the 900 gpm flow rate and would be correct IF the vessel water level were < 23' above the flange.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since this is the correct Tech Spec basis IF the vessel water level were < 23' above the flange.

APE: 025 Loss of Residual Heat Removal System (RHRS)

025 AA2.05 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
(CFR: 43.5 / 45.13)

Importance Rating:	3.1 3.5
Technical Reference:	Tech Spec 3.9.8.1 and Basis
References to be provided:	None
Learning Objective:	RHR Objective 12.f
Question Origin:	New
Comments:	None
Tier/Group:	T1G1
SRO Justification:	Requires application of required Tech Spec below the line application of required actions considered to be SRO knowledge level Tech Spec items and the basis for the actions.

77. Given the following plant conditions:
    - An ATWS has occurred
    - The CRS has transitioned to FR-S.1, Response to Nuclear Power Generation/ATWS
    - The OAC determines that the following occurs in rapid succession:
        - RCS temperature and pressure are rising
        - PZR PORVs indicate OPEN
        - PRT temperature, level and pressure are rising

    Which ONE of the following has occurred AND which procedural action is required?

    A. The Turbine has tripped. Return to E-0

    B. The Turbine has tripped. Continue with FR-S.1

    C. The Turbine Driven AFW Pump has tripped. Transition to FR-H.1

    D. The Turbine Driven AFW Pump has tripped. Continue with FR-S.1

Plausibility and Answer Analysis

Reason answer is correct: *In accordance with the WOG, the indications provided in this question are expected plant responses associated with an ATWS and large load rejection as in a Turbine trip. (RCS temp and pressure excursion, - more severe at BOL - PZR PORVs are expected to open - relieving to the PRT which in turn will cause PRT temp, level and pressure to increase). Indications that the Main Turbine has tripped does not constitue exiting FR-S.1. The crew should remain in FR-S.1 until AFTER the Reactor is determined to be subcritcal which it currently is not.*

A. Incorrect. Indications support Turbine trip response, but FR-S.1 is still in effect due to rising RCS Temp and Press indicating that the Reactor is still acting as a heat source.

B. Correct.

C. Incorrect. TDAFW trip would have effect but MDAFW still available, so not severe enough to cause indications listed. Transition to FR-H.1 is incorrect.

D. Incorrect. TDAFW trip would have effect but MDAFW still available, so not severe enough to cause indications listed. FR-S.1 is still in effect.

EPE : 029 Anticipated Transient Without Scram (ATWS)

029 EA2.09 Ability to determine or interpret the following as they apply to a ATWS: Occurrence of a main turbine/reactor trip
(CFR 43.5 / 45.13)

Importance Rating:	4.4 4.5
Technical Reference:	WOG Background information on FR-S.1 (ATWS Events)
References to be provided:	None
Learning Objective:	EOP-LP-3.2 Objective 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T1G1
SRO Justification:	10 CFR Part 55 Content - 43.5 because the SRO must assess plant conditions and determine a course of action by prescribing a procedure to mitigate the accident.

78. Given the following plant conditions:

- The plant was operating at 100%
- 0600, 'C' SG develops a 15 gpm tube leak and CRS directs a plant shutdown in accordance with AOP-016, Excessive Primary Plant Leakage
- 0610, 1MS-45, MS Line 'C' Safety relief valve, opens and cannot be shut
- 0630, 'C' SG tube leakage degrades and a Reactor Trip and Safety Injection are initiated
- 0645, Chemistry confirms an offsite release to is in progress

Which ONE of the following identifies (1) the FIRST required classification for the conditions above AND (2) the time the State and Counties must be notified ~~of~~ an emergency release in progress?

(Reference provided)

A. (1) FU1.1

(2) 0625

B. (1) FU1.1

(2) 0700

C. (1) SU8.1

(2) 0625

D. (1) SU8.1

(2) 0700

Plausibility and Answer Analysis

Reason answer is correct: An emergency classification is met due to Primary to Secondary leakage greater than 10 gpm with an unisolable leak on the associated generator. This classification starts the 15 minute clock on notifying the State and County officials.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible if candidate believes that a release in progress begins when chemistry has confirmation and start the 15 minute clock on notifying State and County officials.

C. Incorrect. Plausible if candidate believes that Primary to Secondary leakage is classified as unidentified leakage per EP-EAL. The second part is correct.

D. Incorrect. The first part is plausible if candidate believes that Primary to Secondary leakage is classified as unidentified leakage per EP-EAL. The second part is plausible if candidate believes that a release in progress begins when chemistry has confirmation and start the 15 minute clock on notifying State and County officials.

EPE: 038 Steam Generator Tube Rupture (SGTR)

038 EG2.4.30 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)

Importance Rating:	2.7 4.1
Technical Reference:	PEP-310 Rev. 28, page 31 EAL Matrix, S8, and FPBM Table-1 and 2
References to be provided:	None
Learning Objective:	LP-PP-2.17 Obj. 5 & 6.c
Question Origin:	New
Comments:	(K/A match) This question requires the candidate to apply Reportability requirements for PEP-310 for a SG Tube failure.
Tier/Group:	T1G1
SRO justification:	10 CFR Part 55 Content - 43.5 because the SRO must make an assessment of EAL classification and notifications to an outside organization, i.e. the state and counties.

79. Given the following plant conditions:
    - The plant is operating at 86% power.
    - The following alarms are received:
        - ALB-019 /5-3B "Cndbstr Pump B Bkr Trip/Trbl"
        - ALB-020 /2-2, Turbine Runback Operative

    - Subsequently the following alarms are received:
        - ALB-026 /1-4, Annun Sys 1 Power Supply Failure
        - ALB-003 / 4-5, Annunciator System 2 Power Supply Failures

    - The OAC reports 23 of the 30 Main Control Board ALBs have lost annunciators
    - The AO reports the both Annuciator System 1 and Annunciator System 2 have lost multiple 125 VDC power supplies

Which ONE of the following identifies the required classification for the conditions above?

(Reference provided)

A. Site Area Emergency, SS2.1

B. Site Area Emergency, SS5.1

C. Alert, SA5.1

D. Unusual Event, SU5.1

Plausibility and Answer Analysis

Reason answer is correct: The trip of a Condensate Booster pump will result in the trip of the same train Main Feed pump (MFP). The trip of a MFP above 60% turbine load will result in a turbine runback to less than 507.2 psig 1st stage pressure. A turbine runback of greater than 25% reactor power is a significant transient per Table S-1. A significant transient concurrent with a loss of 75% MCB annunciation meets the threshold for an Alert per SA5.1.

A. Incorrect. Plausible due to the report of the loss of 125VDC power supplies to the annunciator cabinets. The candidate could believe that the annunciator cabinets are supplied by DP-1A-SA or DP-1B-SB.

B. Incorrect. Plausible if the candidate believes ERFIS and OSI-PI are powered from the same supply as the annunciator cabinets.

C. Correct.

D. Incorrect. Plausible if the candidate does not recognize that a turbine runback has occurred and reactor power has lowered more than 25%.

QUESTIONS REPORT

for 2013 NRC SRO REV 5 Written Exam Submittal

APE: 054 Loss of Main Feedwater (MFW)

054 AG2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.
(CFR: 41.10 / 43.5 / 45.12)

Importance Rating:	4.2 4.2
Technical Reference:	AOP-010
References to be provided:	EAL Matrix, S2 and S5
Learning Objective:	CFW Student Text, Obj 11
Question Origin:	New
Comments:	None
Tier/Group:	T1G1
SRO Justification:	10 CFR Part 55 Content - 43.5 because the SRO must assess plant conditions and determine an EAL classification.

80. Given the following plant conditions:
- The unit was operating at 100% power when a loss of Offsite power occurred
- 6.9 KV Emergency Bus 1B-SB 86 lockout accuates
- EDG 'A' fails to start
- The ASI system is supplying RCP seal injection
- RCS temperature is 557°F, and steady
- RCS pressure is 2115 psig, and lowering
- PZR level is 25%, and lowering
- All radiation monitors are normal

The following parameters are observed on the Steam Generators:

	'A' SG	'B' SG	'C' SG
AFW Flow	100 kpph	15 kpph	200 kpph
NR Level	25% rising	35% rising	20% rising

Which ONE of the following is (1) the required procedure transition, if any, at this time AND (2) what would be the required action from the procedure?

A. (1) GO TO E-3, Steam Generator Tube Rupture

(2) Isolate AFW flow into 'B' Steam Generator and shut 1MS-70

B. (1) GO TO E-3, Steam Generator Tube Rupture

(2) Request Chemistry to sample the 1B SG for activity

C. (1) Remain in ECA-0.0, Loss of All AC Power

(2) Isolate AFW flow into 'B' Steam Generator and shut 1MS-70

D. (1) Remain in ECA-0.0, Loss of All AC Power

(2) Request Chemistry to sample the 1B SG for activity

Reason answer is correct: With RCS temperature steady, RCS pressure and PZR Level should be steady as well. Since RCS Pressure and PZR Level are steadily lowering, inventory is being lost. Since all radiation monitors are normal, it can be assumed that no leakage is going into the Containment, or other associated systems. The radiation monitors designed to identify a SGTR are either isolated (i.e. MSIVs are closed), or require long periods of time to register a change. With AFW flow into B SG substantially reduced compared to that of any other, the B SG NR level should NOT be increasing. In ECA-0.0 step 18, the operator is directed to check SG, and MSL radiation or SG activity sample normal. All of these indications are considered normal based of the given information. Step 19 checks SG levels and if any are rising in an uncontrolled manner. If indications are rising uncontrolled, according to the Step 19.d, the operator will isolate AFW flow into a ruptured S/G, raise the setpoint for the SG PORV to 88% and isolate Steam flow to the TD AFW pump from the ruptured SG. Steam flow must be maintained to the TD AFW Pump from at least one S/G. The C S/G remains available to supply steam to the TD AFW pump as required.

A. Incorrect. The first part is plausible because the indications are that of a classic SGTR symptom while in the EOP network would normally require a transition to E-3. However, according to the EOP User's Guide (Rev 39, pg 6), if at any time a complete loss of power on the AC Emergency Buses takes place, the operator will enter ECA-0.0, Loss Of All AC Power. This includes any time during the performance of any other EOP, i.e., the implementation of ECA-0.0, always takes precedence under these plant conditions. The second part is correct.

B. Incorrect. The first part is plausible because the indications are that of a classic SGTR symptom while in the EOP network would normally require a transition to E-3. However, according to the EOP User's Guide (Rev 39, pg 6), if at any time a complete loss of power on the AC Emergency Buses takes place, the operator will enter ECA-0.0, Loss Of All AC Power. This includes any time during the performance of any other EOP, i.e., the implementation of ECA-0.0, always takes precedence under these plant conditions. The second part is plausible because the RNO action for the SGTR identification step of E-3 is to coordinate with plant staff to obtain primary and secondary samples to confirm the ruptured SG because radiation alarms are normal..

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible becuase the RNO action for the SGTR identification step of E-3 is to coordinate with plant staff to obtain primary and secondary samples to confirm the ruptured SG because radiation alarms are normal.

EPE: 055 Loss of Offsite and Onsite Power (Station Blackout)

055 EA2.04 Ability to determine or interpret the following as they apply to a Station Blackout: Instruments and controls operable with only dc battery power available
(CFR 43.5 / 45.13)

Importance Rating:	3.7 4.1
Technical Reference:	ECA-0.0, Rev 1, pg 26,28 EOP User's Guide, Rev 41, pg 12
References to be provided:	None
Learning Objective:	EOP-LP-3.7 Obj 6
Question Origin:	Bank
Comments:	None
Tier/Group:	T1G1
SRO Justification:	SRO-Only because the question requires that the operator assess plant conditions, and then prescribe a procedure or a portion of the procedure (i.e. ECA-0.0), including when that strategy is required (i.e. not to transition to E-3 for SGTR when within ECA-0.0).

81. Given the following plant conditions:
    - The Unit is operating at 15% power.
    - A ground on DP-1B-SB results in a loss of the bus.
    - The crew has entered AOP-025, Loss Of One Emergency AC Bus (6.9KV) Or One Emergency DC Bus (125V)
    - The OAC and BOP are monitoring the MCB indications associated with the loss of DP-1B-SB.

    Which ONE of the following completes the statement below for the given conditions?

    During an accident, radiation monitor ___(1)___ is used in determining an event classification in accordance with EP-EAL, Emergency Action Levels, AND ___(2)___ be available.

    A. (1) RM-1CR-3590-SB, Containment High Range Monitor

       (2) will

    B. (1) RM-1MS-3592-SB, Main Steam Line B

       (2) will

    C. (1) RM-1CR-3590-SB, Containment High Range Monitor

       (2) will NOT

    D. (1) RM-1MS-3592-SB, Main Steam Line B

       (2) will NOT

Plausibility and Answer Analysis

Reason answer is correct: Technical Specification 3.3.3.6 Accident monitoring instrumentation instrument 20 identifies the Containment-High Range Radiation Monitor as a required component for satisfying its LCO. OST-1020 additionally list this radiation monitor as a required monitor to satisfy surveillance 4.3.3.6. OP-118 idenifies the power supply for RM-1CR-3590B-SB as PP-1B211-SB cubicle 31 which is an AC power supply and will not be affected by the loss of DP-1B-SB. EP-EAL uses RM-1CR-3590B-SB twice in Table F-1, Fission Product Barrier Matrix, once under the Fuel Clad barrier under Loss and once under the Containment barrier for a Potential Loss determination.

A. Correct.

B. Incorrect. The first part is plausible if the candidate believes that RM-1MS-3592-SB is used in the EAL matrix to classify an event for offsite or onsite radiological conditions. The second part is correct because RM-1MS-3592-SB is powered by 1DP-1B-SIV-11 per OP-118. This monitor would be available with power from the AC source through the channel 4 inverter.

C. Incorrect. The first part is correct. The second is plausible if the candidate believes RM-1CR-3590-SB is powered by DP-1B-SB.

D. Incorrect. The first part is plausible if the candidate believes that RM-1MS-3592-SB is used in the EAL matrix to classify an event for offsite or onsite radiological conditions. In the second part, RM-1MS-3592-SB is powered by 1DP-1B-SIV-11 per OP-118 and is plausible if the candidate believes monitor would be available due to the loss of DC power only.

APE: 058 Loss of DC Power

058 AG2.4.3 Ability to identify post-accident instrumentation.
(CFR: 41.6 / 45.4)

Importance Rating:	3.7 3.9
Technical Reference:	OST-1020, Rev 23, Pg 11 OP-118, Rev 29, Pg 48 EP-EAL, Rev 10, Pg 245
References to be provided:	None
Learning Objective:	RMS Student Text Obj 9.c
Question Origin:	New
Comments:	None
Tier/Group:	T1G1
SRO Justification:	SRO because the determination of the use of the instrument in the Radiological Emergency Plan implementation must be determined.

82. Given the following plant conditions:
    - The unit is operating at 100% power
    - A load rejection occurs which reduces power to 80%
    - The OAC reports that two control rods in Bank D are misaligned by approximately 20 steps
    - The AO dispatched to investigate reports no blown fuses or obvious electrical problems at the Rod Control Cabinets
    - All rods have been verified above the Rod Insertion limits
    - BOTH rods have now been determined to be stuck and immovable

    Which ONE of the following actions are required in accordance with Technical Specifications and AOP-001, Malfunction of Rod Control and Indication System?

    Check SDM within limits of the COLR within one hour and...

    A. place the unit in MODE 3 within 6 hours.

    B. restore the inoperable rods to OPERABLE within 8 hours or open the Reactor Trip System breakers.

    C. power may be maintained at the current value if a power distribution map based on movable incore detectors is verified to be within limits.

    D. power may be maintained at the current value if reevaluation of accident analysis results remain valid for the current condition.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with Tech Spec 3.1.3.1 Action a - with one or more rods inoperable due to being immovable as a result of excessive friction or mechanical interference or know to be untrippable (there are 2 rods with this condition in this question), determine that the SHUTDOWN MARGIN requirement of Spec 3.1.1.1. is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

A. Correct.

B. Incorrect. Plausible since this is the required Tech Spec action for rod position indication when it is not capable of determining the rod position within ± 12 steps for each shutdown or control rod not fully inserted. (This spec applies in MODES 3, 4 and 5)

C. Incorrect. Plausible since this is the action associated with one rod trippable but inoperable due to causes other than addressed by action a (inoperable or immovable rod).see LCO 3.1.3.1 Action d.3.c).

D. Incorrect. Plausible since this is the action associated with one rod trippable but inoperable due to causes other than addressed by action a (inoperable or immovable rod) see LCO 3.1.3.1 Action d.3.a).

APE: 005 Inoperable/Stuck Control Rod / 1

005 AA2.03 Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable
(CFR: 43.5 / 45.13)

Importance Rating:	3.5 4.4
Technical Reference:	Technical Specification 3.1.3.1 Action a. AOP-001, Attachment 5, Rev. 40
References to be provided:	None
Learning Objective:	RODCS Objective 15 (SRO ONLY)
Question Origin:	Bank
Comments:	None
Tier/Group:	T1G2
SRO Justification:	Requires application of required Tech Spec below the line application of required actions considered to be SRO knowledge level Tech Spec items.

83. Given the following plant conditions:
    - The unit is operating at 100% power.
    - RM-1WV-3546-1, Waste Process Building Vent Stack 5 (WRGM), is Inoperable
    - A pre-release Waste Gas Decay Tank release log is being completed for a release planned for this shift.
    - While documenting the flow rate of WPB Vent Stack 5, the operator notes that REM-3546 PIG (4GG793), readings are light blue (cyan) on the RM-11 console.
    - The RM-11 status screen for REM-3546 indicates per the reference provided

    Which ONE of the following (1) decribes the status of REM-3546 PIG (4GG793), AND (2) what actions are required to continue with the Waste Gas Decay Tank release?

    (Reference provided)

    A. (1) Operable - communication with RM-11 is lost, but does not prevent auto actuation from a REM-3546 signal.

       (2) Release via this pathway may continue provided grab samples are taken at least once per 12 hours and analyzed within 24 hours.

    B. (1) Operable - communication with RM-11 is lost, but does not prevent auto actuation from a REM-3546 signal.

       (2) Releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

    C. (1) Inoperable - equipment failure is present for REM-3546.

       (2) Releases via this pathway may continue provided grab samples are taken at least once per 12 hours and analyzed within 24 hours.

    D. (1) Inoperable - equipment failure is present for REM-3546.

       (2) Releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

means

Plausibility and Answer Analysis

Reason answer is correct: ODCM 3.3.3.11 requires each radioactive gaseous effluent monitoring channel to be operable by perfoming a channel check, source check, channel calibration and channel operational test. The RM-11 reading for REM-3546 reading cyan the rad monitor has an equipment failure and is not able to pass the checks and test required to be operable. The ODCM actions for an inoperable radiation monitor are to perform Action 45 and 51. Action 51 allows the release of radioactive effluent via the inoperable pathway provided grab samples are taken once per 12 hours and analyzed for activity within 24 hours.

A. Incorrect. The first part is plausible because channel checks for a non-effluent rad monitor may be performed at the respective RM-23 locally to determine operability status of the rad monitor. The second part is the correct action.

B. Incorrect. The first part is plausible because channel checks for a non-effluent rad monitor may be performed at the respective RM-23 locally to determine operability status of the rad monitor. The second part is plausible because it is the action required to be performed if the flow rate or sampler flow rate monitor were the component in the WPB Vent Stack 5 with a cyan indication for its readings.

C. Correct.

D. Incorrect. The first part is the correct status of operability. The second part is plausible because it is the action required to be performed if the flow rate or sampler flow rate monitor were the component in the WPB Vent Stack 5 with a cyan indication for its readings

APE: 060 Accidental Gaseous Radwaste Release / 9

060 AG2.2.37 Ability to determine operability and/or availability of safety related equipment.
(CFR: 41.7 / 43.5 / 45.12)

Importance Rating:	3.6 4.6
Technical Reference:	OP-118 Rev 29, pg 56, 59 ODCM 3.3.3.11 Rev 6, pg D-7, 8, 9
References to be provided:	ODCM 3.3.3.11 Rev 6, pg D-7, 8, 9 RM-11 screen print
Learning Objective:	RMS Student Text Obj 11
Question Origin:	New
Comments:	None
Tier/Group:	T1G2
SRO Justification:	The question requires the applicant to apply tech specs in the second half of the question. SRO-only linked to 10CFR55.43(b)(2)

84. Given the following plant conditions:
    - The unit is operating at 100% power
    - RM-01CR-3561C-SA, Containment Ventilation Isolation (CVI) Radiation Monitor failed 3 days ago and OWP-RM-02 has been implemented
    - The BOP is performing the Technical Specification required channel check on the remaining operable CVI Radiation Monitors

    Which ONE of the following identifies the normal indication for CVI Radiation Monitors and the Technical Specification required action if another CVI Radiation Monitor fails its channel check?

	Normal Indication	Required Action
A.	110 mRem/hour	CLOSE the CNMT Purge Makeup and Exhaust Isolation Valves
B.	110 mRem/hour	Restore ONE monitor within 7 days or submit a Special Report
C.	$4.8E^{-06}$ μCi/ml	CLOSE the CNMT Purge Makeup and Exhaust Isolation Valves
D.	$4.8E^{-06}$ μCi/ml	Restore ONE monitor within 7 days or submit a Special Report

Plausibility and Answer Analysis

Reason answer is correct: 110 mRem/hour is the normal reading for the RM-3561s, CNMT Ventilation Isolation Rad Monitors at 100% power. Per Technical Specification Table 3.3-6 action 27 - CLOSE the CNMT Purge Makeup and Exhaust Isolation Valves is the action for when two RM-3561s, CNMT Ventilation Isolation Rad Monitors are inoperable.

A. *Correct.*

B. *Incorrect. 110 mRem/hour is the normal reading for the RM-3561s, CNMT Ventilation Isolation Rad Monitors at 100% power. Second part is plausible because the 7 days or special report per Technical Specification 3.3.3.6 action c pertains to another Containment radiation monitor - the High Range Radiation Monitors.*

C. *Incorrect.* $4.8E^{-06}$ μCi/ml *is the normal indication for RM-3502A, CNMT Leak Detection Rad Monitor. Second part is plausible because the 7 days or special report per Technical Specification 3.3.3.6 action c pertains to another Containment radiation monitor - the High Range Radiation Monitors.*

D. *Incorrect.* $4.8E^{-06}$ μCi/ml *is the normal indication for RM-3502A, CNMT Leak Detection Rad Monitor. Per Technical Specification Table 3.3-6 action 27 - CLOSE the CNMT Purge Makeup and Exhaust Isolation Valves is plausible because this is the action for when two RM-3561s, CNMT Ventilation Isolation Rad Monitors are inoperable.*

APE: 061 Area Radiation Monitoring (ARM) System Alarms / 7

061 AA2.02 Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Normal radiation intensity for each ARM system channel
(CFR: 43.5 / 45.13)

Importance Rating:	2.9 3.7
Technical Reference:	OWP-RM-02 Rev. 37 page 10, Technical Specification 3.3.3.1 Table 3.3-6 action 27
References to be provided:	None
Learning Objective:	Student Text Radiation Monitoring System, Obj. 11
Question Origin:	Bank
Comments:	(K/A match) Candidate must recall the normal reading for RM-01CR-3561's and apply appropriate actions to meet Technical Specification requirements.
Tier/Group:	T1G2
SRO Justification:	Requires knowledge of a Technical Specification action that is greater than 1 hour

85. Given the following plant conditions:
    - At 0900 on September 1st while performing EST-219, Personnel Air Lock (PAL) Door Seals Local Leak Rate Test, the Inner door seal fails
    - At 0800 on September 3rd while repairs are in progress on the PAL Inner door seal the Emergency Air Lock (EAL) Outer door sealing surface is damaged by a scaffold pole

    Which ONE of the following completes the statement below?

    The EAL may be entered under administrative controls until ___(1)___ AND during this period of time the performance of repairs to non-vital plant equipment is ___(2)___ in Containment.

    (Reference provided)

    A. (1) 0900 on September 8th

       (2) allowed

    B. (1) 0900 on September 8th

       (2) not allowed

    C. (1) 0800 on September 10th

       (2) allowed

    D. (1) 0800 on September 10th

       (2) not allowed

Plausibility and Answer Analysis

Reason the answer is correct: Per T.S. 3.6.1.3, action a is applicable because one door in each air lock is inoperable. The EAL may be entered under administrative controls for 7 days if BOTH air locks are inoperable per the note (3.6.1.3.a.#.2). The basis section of 3.6.1.3 states that the note is not intended to preclude performing other activities (i.e. non-TS required activies or repairs on non-vital plant equipment.)

A. Incorrect. The first part is plausible if the candidate believes that entry and exit is permissible for 7 days based on the inoperability of the PAL inner door. The second part is correct.

B. Incorrect. The first part is plausible if the candidate believes that entry and exit is permissible for 7 days based on the inoperability of the PAL inner door. The second part is plausible if the candidate believes that entry and exit is only permissible to perform repairs on the affected air lock.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible if the candidate believes that entry and exit is only permissible to perform repairs on the affected air lock.

APE: 069 Loss of Containment Integrity

069 AG2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
(CFR: 41.5 / 41.7 / 43.2)

Importance Rating:	3.2 4.2
Technical Reference:	Technical Specification 3.6.1
References to be provided:	Technical Specification 3.6.1.3
Learning Objective:	CONT Student Text Obj 13.f
Question Origin:	New
Comments:	None
Tier/Group:	T1G2
SRO Justification:	Facility operating limitations in the technical specifications and their bases [10CFR55.43(b)(2)] Knowledge of tech spec bases that is required to analyze tech spec required actions and terminology

86. Given the following plant conditions:
    - 'B' Main Steam Line radiation monitor is in HIGH alarm
    - The crew trip the Reactor and actuate Safety Injection
    - Both MDAFW pumps are unavailable due to common cause motor problems
    - After the Reactor Trip, one 'B' SG safety valve stuck open
    - MSIV's will not close

Which ONE of the following completes the statement below?

1MS-70, Main Steam B To Aux Fw Turbine, is required to be ___(1)___ AND ___(2)___ will be used to ensure 1MS-70 is in the required position.

A. (1) maintained open

(2) EOP-E-2, Faulted Steam Generator Isolation

B. (1) maintained closed

(2) EOP-E-2, Faulted Steam Generator Isolation

C. (1) maintained open

(2) E-3, Steam Generator Tube Rupture

D. (1) maintained closed

(2) E-3, Steam Generator Tube Rupture

Plausibility and Answer Analysis

Given the current plant conditions, there will be no AFW source other than the TDAFW pump. TDAFW steam supply valves will be addressed in multiple procedures for this scenario. EOP-E-2 will be entered from E-0, based on lowering SG pressure. EOP-E-2 will transition to EOP-ECA-2.1 based on all SG pressures going down. The first step in EOP-ECA-2.1 directs foldouts to be implemented, at which time the foldout for E-3 transition will apply based on high radiation. E-3 directs isolating 1MS-70 as long as another supply is available.

A. Incorrect — *Plausible as the TDAFW pump is the only source of AFW available. The second part is plausible because EOP-E-2 will isolate a faulted SG by shutting the faulted SG steam supply valve to the TDAFW pump. With the conditions given in this question this step would not be performed because the procedure will be exited prior to reaching the actions to close the valve. In step 4 of EOP-E-2 the check any SG pressure - Stable Or Rising (not faulted) is performed. Since one 'B' SG safety valve is stuck open and ALL MSIV's will not close ALL SGs are faulted and all SG pressures would be decreasing. Therefore a transition to ECA-2.1 would be made prior to reaching step 6 to isolate 1MS-70.*

B. Incorrect *First part is correct. The second part is plausible because EOP-E-2 will isolate a faulted SG by shutting the faulted SG steam supply valve to the TDAFW pump. With the conditions given in this question this step would not be performed because the procedure will be exited prior to reaching the actions to close the valve. In step 4 of EOP-E-2 the check any SG pressure - Stable Or Rising (not faulted) is performed. Since one 'B' SG safety valve is stuck open and ALL MSIV's will not close ALL SGs are faulted and all SG pressures would be decreasing. Therefore a transition to ECA-2.1 would be made prior to reaching step 6 to isolate 1MS-70.*

C. Incorrect *First part is plausible as the TDAFW pump is the only source of AFW available. Second part is correct.*

D. Correct

QUESTIONS REPORT
for Final Copy 2011 HNP NRC SRO QUESTIONS

2011 NRC SRO 013

Given the following intitial plant conditions:
- 'B' Main Steam Line radiation monitor is in HIGH alarm
- The operating crew initiated a Reactor trip and Safety Injection
- Both MDAFW pumps are unavailable due to common cause motor problems

- After the Reactor Trip, one 'B' SG safety valve stuck open
- MSIV's will not close

Which ONE of the following describes:

(1) The required operation of 1MS-70, MAIN STEAM B TO AUX FW TURBINE

(2) Which procedure will be used to close 1MS-70

A. (1) maintained open

(2) EOP-EPP-014, Faulted Steam Generator Isolation

B. (1) maintained closed

(2) EOP-EPP-014, Faulted Steam Generator Isolation

C. (1) maintained open

(2) PATH-2

D✓ (1) maintained closed

(2) PATH-2

039 A2.03 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)

Importance Rating:	3.4 3.7
Technical Reference:	EOP-E-2, rev. 0, pg 3 EOP-ECA-2.1, rev. 0, pg 3 EOP-E-3, rev. 0, pg 8
References to be provided:	None
Learning Objective:	EOP-LP-3.2, Obj 4.b
Question Origin:	Bank
Comments:	Previous 2011 NRC SRO 13
Tier/Group:	T2G1
SRO Justification:	Requires assessing plant conditions and selecting a procedure sequence with which to proceed.

87. Given the following plant conditions:
    - The unit is operating at 100% power
    - A small break LOCA occurrs
    - The crew tripped the Reactor and actuated Safety Injection
    - RCS pressure is 1175 psig and slowly lowering
    - The crew is implementing E-1, Loss of Reactor or Secondary Coolant
    - Safety Injection has been reset
    - A Grid distrubance results in the loss of offsite power

    Which ONE of the following (1) identifies the transition from E-1, required by the CRS AND (2) the E-0, Reactor Trip or Safety Injection, attachment used to verify proper configuration of safeguards equipment following the loss of offsite power?

    A. (1) ES-1.3, Transfer To Cold Leg Recirculation, Step 1

       (2) Attachment 6, Safeguards Equipment Realignment Following A Loss Of Offsite Power

    B. (1) ES-1.2, Post LOCA Cooldown And Depressurization, Step 1

       (2) Attachment 6, Safeguards Equipment Realignment Following A Loss Of Offsite Power

    C. (1) ES-1.3, Transfer To Cold Leg Recirculation, Step 1

       (2) Attachment 8, Response To Loss of Offsite Power to AC Emergency After SI Actuation

    D. (1) ES-1.2, Post LOCA Cooldown And Depressurization, Step 1

       (2) Attachment 8, Response To Loss of Offsite Power to AC Emergency After SI Actuation

Plausibility and Answer Analysis

Reason answer is correct: The loss of offsite power causes the DG output breaker to automatically re-energize the emergency buses. Following the reset of safety injection the loss of offsite power results in the safeguards sequencer running in program A (UV mode) which does not restart the previously running program C (SI mode) loads, E-0 Attachment 6 provides the list of equipment that must be evaluated for this condition. A small break LOCA will stabilize at an RCS pressure higher than the RHR injection pressure (RCS pressure less than 230 psig) and the CRS should continue in E-1 to step 13 then transition from E-1 to ES-1.2

A. Incorrect. The first part is plausible because the RCS pressure continues to lower however the conditions for a small break LOCA are expected to stablizes above the pressure for injection of the high volume RHR system and the inventory of the RWST should support the cooldown and depressurization of the RCS. The second part is the correct answer.

B. Correct.

C. Incorrect. The first part is plausible because the RCS pressure continues to lower however the conditions for a small break LOCA are expected to stablizes above the pressure for injection of the high volume RHR system and the inventory of the RWST should support the cooldown and depressurization of the RCS. The second part is plausible because Attachment 8 provides the direction to realign equipment following the loss of offsite power, but assumes the EDG of one emergency bus fails to restore power to its associated bus.

D. Incorrect. The first part is the correct answer. The second part is plausible because Attachment 8 provides the direction to realign equipment following the loss of offsite power, but assumes the EDG of one emergency bus fails to restore power to its associated bus.

062 AC Electrical Distribution System

062 A2.11 Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Aligning standby equipment with correct emergency power source (D/G)
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.7 4.1
Technical Reference:	E-1, Rev 0, pg 14, 16 E-0, Rev 0, pg 65
References to be provided:	None
Learning Objective:	EOP-LP-3.1, Obj 5.c
Question Origin:	New
Comments:	None
Tier/Group:	T2G1
SRO Justification:	Assessing plant conditions (normal, abnormal, or emergency) and then prescribing a procedure or section of a procedure to mitigate, recover, etc.

88. Given the following plant conditions:
    - The unit is operating at 100% power
    - An electrician performing the weekly maintenance surveillance test for the 1A-SA Emergency Battery has just informed you of the following pilot cell indications:
        - electrolyte level is midway between the minimum and maximum marks
        - float voltage is 2.10 volts
        - specific gravity is 1.198

    In accordance with Technical Specification 3.8.2.1, D. C. Sources (refer to attached Technical Specifications) the 1A-SA battery is...

    (Reference provided)

    A. operable provided that all Category B parameters are verified within allowable values within 24 hours and Category A and B parameters are restored to within limits within the next 6 days.

    B. inoperable, and must be restored to operable status within 2 hours or the plant must be placed in Mode 3 within the next 6 hours and in Mode 5 within the following 30 hours.

    C. operable provided that all Category A and B parameters are restored to within limits within the next 7 days.

    D. operable provided all Category B parameters are verified to be within their allowable values within 7 days.

Plausibility and Answer Analysis

Reason answer is correct: Per T.S. 4.8.2.1.a.1, the parameters of Table 4.8-2 must meet the Catagory A limits. For a pilot cell in Catagory A, note 1 states, "for any Catagory A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all Catagory B measurements are taken and found to be within their allowable values, and provided all Catagory A and B parameter(s) are restored to within limits within the next 6 days.

A. *Correct.*

B. *Incorrect. Plausible if the candidate believes the reading taken is a connected cell and would fall under Catagory B and is outside the limit for float voltage and specific gravity. Note 3 could be used to determine that the battery is inoperable.*

C. *Incorrect. Plausible if the candidate believes that the pilot cell is connected to the battery and would therefore fall under both Catagory A and B. Note 2 has Catagory B parameters restored to within limits within 7 days.*

D. *Incorrect. Plausible if the candidate beleives that the cell is only a connected cell and would fall under Catagory B. This requires restoration of the parameter within limits within 7 days.*

063 DC Electrical Distribution System

063 G2.2.40 Ability to apply Technical Specifications for a system.
(CFR: 41.10 / 43.2 / 43.5 / 45.3)

Importance Rating:	3.4 4.7
Technical Reference:	ALB-15-5-4, Rev. 21 Technical Specifications 3.8.2.1
References to be provided:	None
Learning Objective:	DC power Student Text, Obj 13.a
Question Origin:	Bank
Comments:	None
Tier/Group:	T2G1
SRO Justification:	Facility operating limitations in the technical specifications and their bases. Application of required actions (Section 3) and surveillance requirements (Section 4) in accordance with rules of application requirements (Section 1)

89. Given the following plant conditions:
    - The unit is operating at 55% power
    - The following annunciator is received in the Control Room:
      ALB-002-7-2, Serv Wtr Pumps Discharge Low Press
    - The BOP notes that Cooling Tower Basin Level is lowering rapidly
    - Service Water header pressure is 45 psig and lowering

    One minute later
    - Service Water header pressure is 25 psig and continues to lower
    - CTMU cannot maintain Cooling Tower Basin level
    - The crew enters AOP-022, Loss of Service Water and completes the immediate actions
    - ALB-002-7-2 is still in and the Cooling Tower Basin Level continues to lower
    - The RAB AO reports that a large volume of water is gushing from the downstream flange of 1SW-276, Headers A & B Return To Normal SW Header valve and the area is inaccessible
    - All automatic plant systems have functioned as designed

    Which ONE of the following describes (1) the location of the leak, and (2) the action required?

    A. (1) Unisolable leak located in the Normal Service Water System

       (2) Trip the Reactor and go to E-0

    B. (1) Unisolable leak located in the Emergency Service Water System

       (2) Trip the Reactor and go to E-0

    C. (1) Isolable leak located in the Normal Service Water System

       (2) Align NSW to equipment listed in AOP-022, Attachment 1

    D. (1) Isolable leak located in the Emergency Service Water System

       (2) Align ESW to equipment listed in AOP-022, Attachment 1

Plausibility and Answer Analysis

Reason answer is correct: *Based on SW pressure indications, ESW should have automatically started and isolated ESW from NSW. Cooling Tower level continuing to decrease therefore indicates a leak on NSW. The report from the field indicates that the leak is nonisolable. Because suction cannot be maintained, and power is >P-10, the Reactor will be tripped.*

A. Correct.

B. Incorrect. The actions are correct but location is incorrect. Plausible if candidate fails to recognize that ESW would have automatically initiated and isolated from NSW in response to the lowering NSW pressure.

C. Incorrect. Plausible since action is correct for an isolable leak, but based on field report of the leak location the leak is not isolable.

D. Incorrect. Plausible since this would be the correct answer if the leak were isolable and on ESW.

076 Service Water System (SWS)

076 A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS (CFR: 41.5 / 43.5 / 45/3 / 45/13)

Importance Rating:	3.5 3.7
Technical Reference:	AOP-022, Section 3.2, Rev 35, pg 30 ALB-002-7-2, Rev. 48 , pg 33
References to be provided:	None
Learning Objective:	AOP-022 Objective 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T2G1
SRO Justification:	Must make an assessment of plant conditions (abnormal, or emergency) and then prescribing a procedure or section of a procedure to mitigate.

90. Given the following plant conditions:
   - The plant is operating at 90% power
   - A steam leak inside Containment results in the following:
      - 1CP-6 & 3 SB, Normal Purge Inlet/Discharge, fails to close
      - TCV97540, ERFIS Containment Average Temperature 122°F
      - PCT9950, ERFIS Containment Average Pressure 1.55 psig

   Which ONE of the following describes (1) the Technical Specification LCO that is not met AND (2) the bases for the applicable action statement?

   (Reference provided)

   A. (1) Technical Specification 3.6.1.5 for Containment Air temperature is not met

      (2) ensures that containment cooling capability will be available in the event of a LOCA or steam line break

   B. (1) Technical Specification 3.6.1.5 for Containment Air temperature is not met

      (2) ensure that the overall containment temperature does not exceed the initial temperature assumed in the safety analysis for a LOCA or steam line break

   C. (1) Technical Specification 3.6.1.7 for Containment ventilation system is not met

      (2) ensures that containment cooling capability will be available in the event of a LOCA or steam line break

   D. (1) Technical Specification 3.6.1.7 for Containment ventilation system is not met

      (2) ensure that the overall containment temperature does not exceed the initial temperature assumed in the safety analysis for a LOCA or steam line break

Plausibility and Answer Analysis

Reason answer is correct: Technical Specification 3.6.1.5 requires the average air temperature of 3 Containment locations to be maintained below 120°F while in Modes 1-4. The action for this LCO is to reduce the average air temperature to below 120°F within 8 hours or to be in Hot Standby within the next 6 hours. The basis section for T.S. 3.6.1.5 states that the limitations "ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a LOCA or steam line break accident.

A. Incorrect. The first part is correct. The second part is plausible because of the correlation between the containment temperature and the required cooling capaibility needed due to a high temperature. However, this is the basis for the Containment Spray system.

B. Correct.

C. Incorrect. The first part is correct because 1CP-6 & 3 are failed open and do not meet the basis of the containment ventilation system. The second part is plausible if the candidate believes the basis of T.S. 3.6.1.7 (Containment Ventilation System) is for containment cooling capability in the event of a LOCA or steam line break.

D. Incorrect. The first part is correct because 1CP-6 & 3 are failed open and do not meet the basis of the containment ventilation system. The second part is plausible if the candidate believes the basis of T.S. 3.6.1.7 (Containment Ventilation System) is so the overall containment temperature does not exceed the initial temperature assumed in the safety analysis.

QUESTIONS REPORT

for 2013 NRC SRO REV 5 Written Exam Submittal

103 Containment System

103 G2.2.22 Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

Importance Rating:	4.0 4.7
Technical Reference:	Technical Specification 3.6.1
References to be provided:	None ??
Learning Objective:	CONT Student Text, Obj 12.f
Question Origin:	New
Comments:	None
Tier/Group:	T2G1
SRO Justification:	Requires knowledge of a Technical Specification action that is greater than 1 hour

91. Given the following plant conditions:
    - The crew is implementing E-1, Loss Of Reator Or Secondary Coolant
    - Plant conditions are as follows:
        - CNMT pressure - 22.6 psig
        - RCS temperature - 650°F, rising slowly
        - Core exit thermocouples
          A08 - 1154°F, rising slowly
          B05 - 1208°F, rising slowly
          G02 - 857°F, rising slowly
          H15 - 753°F, rising slowly
          L14 - 734°F, rising slowly
        - RCS pressure - 200 psig, rising slowly
        - RVLIS Full Range level - 38%
        - Multiple points on the SPTOP display read nan NCAL
        - 3 CSFST displays are WHITE the remaining displays are GREEN

Which ONE of the following identifies (1) the required EAL classification AND (2) the status of the Fission Product Barrier for the classification?

(Reference provided)

A. (1) General Emergency, FG1.1

   (2) Loss of Fuel Clad Barrier, Loss of Reactor Coolant System Barrier AND Potential Loss Of Containment Barrier

B. (1) General Emergency, FG1.1

   (2) Potential Loss of Fuel Clad Barrier, Loss of Reactor Coolant System Barrier AND Potential Loss Of Containment Barrier

C. (1) Site Area Emergency, FS1.1

   (2) Potential Loss of Fuel Clad Barrier AND Loss of Reactor Coolant System Barrier

D. (1) Site Area Emergency, FS1.1

   (2) Loss of Fuel Clad Barrier AND Loss of Reactor Coolant System Barrier

Reason answer is correct: Since multiple points on the SPTOP display are NCAL and several CSFSTs are WHITE, ~~there~~ candidate must determine the status of the Core Cooling tree manually. EOP-USERS-GUIDE states, "The TC value should be considered 'greater than' if five functioning TCs are greater than the setpoint." Therefore, with the conjunction of 5 valid TC reading being greater than 730 degrees and RVLIS full range level being less than 39%, CSFST Core Cooling-RED entry conditions are met in Table F-1, A.1 for the Fuel Clad barrier loss. Subcooling is calculated manually per EOP-USERS-GUIDE by using the highest TC reading and RCS pressure from PI-402A when less than 700 psig. Subcooling is less than 40 degrees and therefore the RCS leak rate is greater than available ECCS makeup capacity per Table F-1, D.2 for the RCS barrier loss. Site Area Emergency, FS1.1 would be classified due to the Loss of Fuel Clad Barrier AND Loss of Reactor Coolant System Barrier.

A. Incorrect. Loss of Fuel Clad and RCS barriers are correct. Plausbile if the candidate believes that the Containment Barrier is potentially lost due to Table F-1, B.3 Containment Potential Loss. Core exit TCs are greater than 730 degrees, RVLIS Full Range level is less than 39%, but EOP-FRP-C.1 has not been entered and effectiveness cannot be determined. This condition would be classified under FG1.1.

B. Incorrect. Plausbile if the candidate believes that the Fuel Clad Barrier is potentially lost due to TCs reading greater than 730 degrees per Table F-1, B.2 Fuel Clad Potential Loss. The candidate could also believe that Containment Barrier is potentially lost due to Table F-1, B.3 Containment Potential Loss. Core exit TCs are greater than 730 degrees, RVLIS Full Range level is less than 39%, but EOP-FRP-C.1 has not been entered and effectiveness cannot be determined. This condition would be classified under FS1.1.

C. Incorrect. Site Area Emergency, FS1.1 is correct. Plausible if the candidate believes that the Fuel Clad Barrier is potentially lost due to TCs reading greater than 730 degrees per Table F-1, B.2 Fuel Clad Potential Loss vice Core Cooling being RED. The Loss of the Reactor Coolant System is correct.

D. Correct.

017 In-Core Temperature Monitor (ITM) System

017 G2.1.7 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)

Importance Rating:	4.4 4.7
Technical Reference:	EAL Matrix, Rev. 10
References to be provided:	EAL Matrix, Fission Product Barrier Matrix, Table F-1, and F-2
Learning Objective:	EP-LP-3.0, rev 0, obj 10
Question Origin:	New
Comments:	None
Tier/Group:	T2G2
SRO Justification:	Recalling what strategy or action is written into a plant procedure, including when the strategy or action is required

92. Given the following plant conditions:
    - At 0200, the unit was operating at 100% power
    - A large break LOCA has occurred with fuel damage that has generated hydrogen into the Containment atmosphere
    - The crew is implementing E-1, Loss of Reactor or Secondary Coolant
    - The hydrogen monitoring system has been aligned

    - At 0400 indications are that Containment hydrogen concentration is currently 0.35% and slowly rising

    - At 0500 Containment hydrogen concentration was 0.52%
    - The CRS directed an AO to place Electric Hydrogen Recombiner 1A in operation in accordance with OP-125, Post Accident Hydrogen Systems

    - At 1800, with 1A Recombiner in operation and 1B Recombiner in standby Containment hydrogen concentration has increased to 3.14%

    Based on these conditions, which ONE of the following actions should be directed to be performed in accordance with OP-125?

    A. Continue operation of ONLY the 1A recombiner until the hydrogen concentration in Containment is below 0.5% then secure the 1A recombiner.

    B. Start the 1B recombiner ONLY after Containment hydrogen concentration has exceeded 4%.

    C. Start the 1B recombiner and operate both recombiners until the Containment hydrogen concentation is below 0.5% then secure BOTH recombiners.

    D. Continue operation of ONLY the 1A recombiner, increase the power out by 4 KW and the frequency of monitoring and logging Containment hydrogen concentration from every 4 hours to hourly.

Plausibility and Answer Analysis

Reason answer is correct: Post LOCA Containment hydrogen concentrations have increased above 0.5%. P&L's 2 and 6 support running BOTH recombiners. After the 1A recombiner has been placed in service the directions should be to place the standby (1B) recombiner in operation.

#2 states Both EHR units should be put in operation when hydrogen concentration exceeds 0.5%.
#6 When the first recombiner has been put in operation the second recombiner should be placed in standby. If hydrogen concentration increases by greater than or equal to 0.5% in 24 hours, the standby recombiner should be placed in service.

A. Incorrect. Plausible since section 7.1 for securing the recombiners initial condition of H2 concentration of <0.5% is met. But since the hydrogen concentration has exceeded 0.5% the requiement is to run both recombiners.

B. Incorrect. Plausible since the 1B recombiner should be started and placed in operation but P&L #1 states that the recombiners hould not be operated if containment H2 concentration is ≥4%.

C. Correct.

D. Incorrect. Plausible since the capability of ONE recombiner during Post LOCA conditions should be sufficient to maintain the H2 concentration to <4%. HNP has two 100% capacity recombiners. The recombiner when in operation should be monitiored hourly in accordance with the startup section 5.1. The frequency of monitoring is therefore NOT increased. Four hours is plausible since this is the time it should take during operation for temperatures to stabilize. Additionally, an increase of Containment H2 concentration of >0.5% would require increasing the power output by 4 KW in accordance with section 5.1 step 4.a.

028 Hydrogen Recombiner and Purge Control System (HRPS)

028 A2.02 Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: LOCA condition and related concern over hydrogen (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.5 3.9
Technical Reference:	E-1, step 23.d RNO, Rev. 0 OP-125, P&Ls and startup section 7.1, Rev. 23
References to be provided:	None
Learning Objective:	PAHC Student Text, Obj 3.e
Question Origin:	New
Comments:	None
Tier/Group:	T2G2
SRO Justification:	Recalling what strategy or action is written into a plant procedure, including when the strategy or action is required

93. Given the following plant conditions:
   - A release of Treated Laundry and Hot Shower Tank 'A' is in progress
   - A HIGH ALARM is received on REM-*1WL-3540, Treated Laundry and Hot Shower Tank Pump discharge radiation monitor
   - Discharge flow indicated on the Waste Processing computer is approximately 28 gpm

   Which ONE of the following is correct concerning the release in progress?

   A. The release must be terminated. Isolate the release path in accordance with AOP-005, Radiation Monitoring.

   B. The release is terminated. Waste Processing computer indication is a setpoint, not actual flow, indicated by the liquid waste release permit. Verify isolation in accordance with AOP-008, Accidental Release of Liquid Waste.

   C. The release may continue because the release permit provides actual sample data of the tank contents. Determine cause of the alarm in accordance with OP-119, Radwaste Radiation Monitoring System.

   D. The release may continue provided that 2 grab samples of the release are taken and analyzed in accordance with CRC-265, Chemistry Control of the Laundry and Hot Shower, Chemical Drains, and Floor Drains Waste Treatment System.

Plausibility and Answer Analysis

Reason answer is correct: *High alarm on an effluent release monitor requires the release to be terminated. In this case, automatic action did not occur, so the action is required manually. With flow indicated, the release is ongoing. Any time the alarm is lit, the release should be stopped.*

A. Correct.

B. Incorrect. Plausible because the release should have automatically terminated but has not since there is indication of actual flow.

C. Incorrect Plausible because actual sample data is obtained for the intial release but the release should have terminated based on the High radiation condition.

D. Incorrect. Plausible since this denotes action for a failed monitor, not an alarming monitor.

QUESTIONS REPORT

for 2013 NRC SRO REV 5 Written Exam Submittal

068 Liquid Radwaste System (LRS)

068 A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation
(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.3 3.3
Technical Reference:	AOP-008, Rev. 14, pg 4, Step 2 RNO AOP-005, Rev. 28, pg 27, Attachment 9 step 4
References to be provided:	None
Learning Objective:	LPAOP3-5 obj 5
Question Origin:	Bank
Comments:	None
Tier/Group:	T2G2
SRO Justification:	Requires assessment of plant conditions and prescribing a procedure or section of a procedure to mitigate the release.

94. Given the following plant conditions:
    - The unit is in Mode 6 and preparations to begin Refueling are in progress

    Which ONE of the following lists the responsibilities of the SRO-Fuel Handling in accordance with PLP-616, Fuel Handling Operations?

    1) Verify that the required Technical Specifications are satisfied prior to fuel movement and once per 6 hours during fuel movement.
    2) Has the authority to stop any action deemed potentially unsafe or detrimental to plant equipment or fuel.
    3) Provide input to the Refueling Coordinator concerning fuel handling activities
    4) Supervise or oversee fuel handling activities in Containment.

    A. 1 and 2

    B. 1 and 3

    C. 2 and 4

    D. 3 and 4

Plausibility and Answer Analysis

Reason answer is correct: PLP-616, the list the responsiblity of the SRO-Fuel Handling as follows:

a. Hold primary responsibility for the safe movement of fuel and core components inside Containment and the Fuel Handling Building.
b. Supervise or oversee fuel handling activities in Containment.
c. Supervise the FHB Operator.
d. Has the authority to stop any action deemed potentially unsafe or detrimental to plant equipment or fuel.
e. Maintain cognizance of fuel handling requirements and limitations.

A. Incorrect. The first item listed is plausible since the SRO-Fuel Handling is a licensed SRO at HNP and has the knowledge to apply Tech Specs to ensure compliance with the refueling specifications. The remaining items are correct.

B. Incorrect. The first item listed is plausible since the SRO-Fuel Handling is a licensed SRO at HNP and has the knowledge to apply Tech Specs to ensure compliance with the refueling specifications. The remaining items are correct.

C. Correct.

D. Incorrect. The first item listed is plausible since the SRO-Fuel Handling is a licensed SRO at HNP and has the knowledge to apply Tech Specs to ensure compliance with the refueling specifications. The remaining items are correct.

2.1 Conduct of Operations

G2.1.41 Knowledge of the refueling process.
(CFR: 41.2 / 41.10 / 43.6 / 45.13)

Importance Rating:	2.8 3.7
Technical Reference:	PLP-616, Rev. 16, pg 11, Section 5.3.2.3
References to be provided:	None
Learning Objective:	LP-PP-2.8 Obj. 2.d
Question Origin:	New
Comments:	None
Tier/Group:	T3
SRO Justification:	Fuel handling facilities and procedures: Responsibilites of SRO licensed individuals specifically the SRO-Fuel Handling and Shift Manager

95. Which ONE of the following choices completes the statement below?

Installation of grounding devices per OPS NGGC-1301, Equipment Clearance, requires that the ground checklist be authorized by a ____(1)____ and documented using a(n) ___(2)___ Verification.

A. (1) Senior Reactor Operator

   (2) Concurrent

B. (1) Senior Reactor Operator

   (2) Independent

C. (1) Electrical Maintenance Supervisor

   (2) Concurrent

D. (1) Electrical Maintenance Supervisor

   (2) Independent

Plausibility and Answer Analysis

Reason answer is correct: *IAW OPS-NGGC-1301 an SRO is required to authorize each checklist (in this case a ground checklist) for installation. Subsection 9.10 step 14 requires concurrent verification that the grounding device is installed on the correct point.*

A. Correct-

B. Incorrect- *Plausible since the first part is correct- A SRO is required to authorize the ground checklist for installation of grounds. The second part is incorrect, but is plausible since independent verification is a human performance tool utilized in some procedures for verification practices.*

C. Incorrect- *An SRO is required to authorize the ground checklist. The distractor is plausible since the Electrical Maintenance Supervisor ensure personnel under their supervision comply with the requirements of this procedure for the use and specification of Grounding Devices (OPS-NGGC-1301 Section 4.5, Supervisor responsibilites). Maintenance personnel are responsible for applying ground devices and the assumption could be made that a supervisor of the cognizant work group would grant authorization for work. The second part is correct, OPS-NGGC-1301 requires concurrent verification that the grounding device is installed on the correct point.*

D. Incorrect- *The first part is incorrect (see C). The second part is incorrect, but is plausible since independent verification is a human performance tool utilized in some procedures for verification practices.*

2.2 Equipment Control

G2.2.13 Knowledge of tagging and clearance procedures.
(CFR: 41.10 / 45.13)

Importance Rating:	4.1 4.3
Technical Reference:	OPS-NGGC-1301, Rev. 30, pg 42, Section 9.9 Step 12, and pg 46, Section 9.10 Step 14
References to be provided:	None
Learning Objective:	Lesson Plan PP2-4 Objective 3
Question Origin:	Bank
Comments:	This question has been changed to reflect the current practices of OPS-NGGC-1000. When originally developed the practice was to perform independent verification for grounding devices. The current practice is to use concurrent verification. Therefore answer A is now correct in comparision to the bank question where answer B was chosen as correct. (AL 2/18/2013)
Tier/Group:	T3
SRO Justification:	Assessment of situational conditions and unique responsibilites of the SRO position.

96. Which ONE of the following completes the statement below?

In accordance with Technical Specifications, at 100% power, shutdown margin must be verified by __(1)__ in order to protect against a __(2)__ .

A. (1) performing a calculation using OST-1036, Shutdown Margin Calculation Modes 1-5

(2) steam line break

B. (1) performing a calculation using OST-1036, Shutdown Margin Calculation Modes 1-5

(2) dilution accident

C. (1) checking rods above rod insertion limits using OST-1021, Daily Surveillance Requirements Mode 1, 2

(2) steam line break

D. (1) checking rods above rod insertion limits using OST-1021, Daily Surveillance Requirements Mode 1, 2

(2) dilution accident

Plausibility and Answer Analysis

Reason answer is correct: Technical Specification. 4.1.1.1.1.b requires verifying rods are above the insertion limits of Technical Specification 3.1.3.6 which is performed in OST-1021. In modes 1 and 2, the most restrictive conditions for shutdown margin are associated with a postulated steam line break and the resulting uncontrolled RCS Cooldown.

A *Incorrect. OST-1036 is plausible because it is used in Modes 3-5 and has a section for performance in Modes 1 and 2 but Technical Specification 4.1.1.1.1.b requires verifying rods are above the insertion limits of Technical Specification 3.1.3.6 which is performed in OST-1021. In modes 1 and 2, the most restrictive conditions for shutdown margin are associated with a postulated steam line break and the resulting uncontrolled RCS Cooldown.*

B *Incorrect. OST-1036 is OST-1036 is plausible because it used in Modes 3-5 and has a section for performance in Modes 1 and 2 but Technical Specification 4.1.1.1.1.b requires verifying rods are above the insertion limits of Technical Specification 3.1.3.6 which is performed in OST-1021. Dilution accident is plausible because this is the basis for Modes 3-5.*

C *Correct.*

D *Incorrect. Technical Specification 4.1.1.1.1.b requires verifying rods are above the insertion limits of Technical Specification 3.1.3.6 which is performed in OST-1021. Dilution accident is plausible because this is the basis for Modes 3-5.*

G2.2.40 Ability to apply Technical Specifications for a system.
(CFR: 41.10 / 43.2 / 43.5 / 45.3)

Importance Rating:	3.4 4.7
Technical Reference:	Technical Specification 3.1.1.1 pg 3/4 1-1 (page 95) Technical Specification Bases pg B 3/4 1-1 (page 384)
References to be provided:	None
Learning Objective:	LP-TS-2.0 Obj. 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T3
SRO Justification:	Requires procedure selection, application of Tech Spec surveillance requirements and bases for the requirements.

97. Given the following plant conditions:
    - During Fuel shuffle a spent fuel assembly was damaged as it was being inserted in a new storage location
    - Bubbles are observed breeching the surface of the Spent Fuel Pool
    - Multiple Fuel Handling Building area radiation monitors and the Plant Vent Stack Radiation Monitor are rising but have NOT reached ALERT or HIGH alarm set points
    - The MCR implemented AOP-013, Fuel Handling Accident
    - While evacuating the Spent Fuel Bridge the operator slips, breaking his leg and tearing his PC's
    - The bridge operator is transported to Western WakeMed for treatment

Which ONE of the following completes the statements below?

In accordance with AOP-013 ___(1)___ is the primary radiological concern for fuel off-loaded more than 6 months ago because it will NOT be detected by personal dosimetry or area radiation monitors.

In accordance with AP-617, Reportability Determination And Notification, the NRC is ___(2)___ to be notified of this event.

A. (1) Krypton-85

   (2) required

B. (1) Krypton-85

   (2) NOT required

C. (1) Iodine-131

   (2) required

D. (1) Iodine-131

   (2) NOT required

Plausibility and Answer Analysis

Reason answer is correct: AOP-013 checks SFP radiation monitors for HIGH ALARM conditions. If they are NOT in HIGH ALARM but radiation levels are rising then step 5 RNO directs starting at least one train of FHB Emergency Exhaust. Step 8 checks all stack radiation monitor levels NORMAL. In this case the stack radiation indication is trending up which requires Chemistry to draw and analyze periodic stack samples and compare any release to the ODCM limits. An SRO would be required to perform an offsite dose calculation IF the stack radiation monitor were in HIGH ALARM.

With the plant operating at 100% power for 17 months all of the fuel in the Spent Fuel Pool has been there for more than 6 months. The personal non-detectable radiation hazard would be Krypton-85 which is a beta emitter. AOP-013 has a note stating: Kr-85 is the primary radiological concern for fuel off-loaded more than 6 months ago. Kr-85 is a beta hazard and will NOT be detected by personal dosimetry or area radiation monitors. There is also a caution stating: Airborne radiation may be present and gas bubbles may be visible if a fuel assembly is ruptured. Personnel should remain clear until Health Physics has established access controls

The basis document states the activity of most concern is that which is contained in the volatile fission product gases contained in the fuel pellet to cladding gap. When a fuel pin is damaged, this fission product inventory can be released to the SFP water. Technical Specifications 3.9.10 and 3.9.11 require a minimum water level of 23 feet in the SFP and Refueling Cavity specifically to reduce the potential dose resulting from a fuel handling accident. This amount of water will capture 99% of the assumed 10% iodine activity present in the pellet to clad gap before it breaks the surface of the water. However, although the water is expected to retain a large fraction of this activity, a portion of it will reach the surface and bubble out into the FHB or CNMT atmosphere. (Since halogens are soluble, a large fraction of these halogens will be retained by the water, whereas noble gases, being insoluble, will not be retained.) Once in the atmosphere, much of this fission product activity will cause an observed increase in area radiation levels. (Gases such as Kr-85 which are primarily beta hazards will not be detectable using installed monitors.)

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since the Plant Vent Stack Radiation Monitor is NOT normal which requires procedure RNO actions. But since it has NOT reached the HIGH ALARM setpoint the action to perform an offsite dose calculation will not be required.

C. Incorrect. The first part is plausible since volatile fission product gases have escaped from the damaged fuel assembly and were seen as observable bubbles coming to the surface of the SFP. Iodine 131 would be part of the volatile gases. I-131 is gamma emiter which would be detectable with personal dosimetry and therefore would NOT be a non-detectable radiation concern. The second part is plausible since the Plant Vent Stack Radiation Monitor is NOT normal which requires procedure RNO actions. But since it has NOT reached the HIGH ALARM setpoint the action to perform an offsite dose calculation will not be required.

D. Incorrect. The first part is plausible since volatile fission product gases have escaped from the damaged fuel assembly and were seen as observable bubbles coming to the surface of the SFP. Iodine 131 would be part of the volatile gases. I-131 is gamma emiter which would be detectable with personal dosimetry and therefore would NOT be a non-detectable radiation concern. The second part is correct.

2.3 Radiation Control

G2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
(CFR: 41.12 / 43.4 / 45.10)

Importance Rating:	3.4 3.8
Technical Reference:	AOP-013-BD Rev. 3 AOP-013 Rev. 15
References to be provided:	None
Learning Objective:	Lesson Plan AOP3-13, Obj 5
Question Origin:	New
Comments:	None
Tier/Group:	T3
SRO Justification:	10CFR55.43(b).5 because the SRO is assessing conditions to determine a course of action. 10CFR55.43(b).6 because these actions are associated with Fuel Handling systems

98. The MAXIMUM dose allowed for life saving missions during a declared emergency is __(1)__ REM TEDE and in accordance with PEP-330, Radiological Consequences, the level of authority that can authorize this dose is the __(2)__ .

A. (1) 10

(2) Site Emergency Coordinator

B. (1) 10

(2) Radiological Control Director

C. (1) 25

(2) Radiological Control Director

D. (1) 25

(2) Site Emergency Coordinator

Plausibility and Answer Analysis

Reason answer is correct: PEP-330, Attachment 1 states that *25 Rem TEDE is the maximum dose allowed for life saving missions during a declared emergency and the lowest level of authority for authorization of the life saving dose can be from the Site Emergency Coordinator (SEC).*

A. Incorrect. 10 Rem TEDE is plausible because this is the limit for protecting valuable equipment but life saving and the limit is 25 Rem TEDE. Site Emergency Coordinator (SEC) is correct.

B. Incorrect. 10 Rem TEDE is plausible because this is the limit for protecting valuable equipment but life saving and the limit is 25 Rem TEDE. The Radiological Control Director (RCD) is plausible because this individual works in the TSC and coordinates the authorization of dose greater than the 10CFR limits with the SEC.

C. Incorrect. 25 Rem TEDE is correct. The Radiological Control Director (RCD) is plausible because this individual works in the TSC and coordinates the authorization of dose greater than the 10CFR limits with the SEC.

D. Correct.

2.3 Radiation Control

G2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)

Importance Rating:	3.2 3.7
Technical Reference:	PEP-330, Rev. 9, pg 17, Attachment 1
References to be provided:	None
Learning Objective:	EP-LP-2.0, SRO Obj 1
Question Origin:	Bank
Comments:	(K/A match) Knowledge of radiation exposure limits during an emergency (in accordance with PEP-330 attachment 1)
Tier/Group:	T3
SRO Justification:	Shift Manager fills the role of the SEC until relieved the the SEC-TSC. SRO must be knowledgable in the event the Shift Manager is unable to report to the MCR during an event.

99. Which ONE of the following lists the non-delegable responsibilities of the SEC-MCR in accordance with PEP-230, Control Room Operations?

    1) Classification and declaration an emergency
    2) Provide protective action recommendations (PARs)
    3) Activation of the NRC ERDS data link within 60 minutes
    4) Complete Accountability within 30 minutes of declaring a Site Area Emergency

A. 1 and 2

B. 1 and 3

C. 2 and 4

D. 3 and 4

Plausibility and Answer Analysis

Reason answer is correct: PEP-230, the list the non-delegable responsiblity of the SEC-MCR as follows:

a. Classification and declaration an emergency
b. Notification of state and counties within 15 minutes of a classification
c. PARs and authorization of emergency exposures to ERO members

A. Correct.

B. Incorrect. The first item listed is correct. The second item listed is plausible because if the activatin of ERDS is perfomed as part of the SEC-MCR checklist for Attachment 1 of PEP-230, but this is not one of the non-delegable responsibilities

C. Incorrect. The first items listed is correct. The second item listed is plausible because if the event classification is an Site Area Emergency the ERO facilities are required to perform accountability within 30, but this is not one of the non-delegable responsibilities.

D. Incorrect. The first tem listed is plausible because if the activatin of ERDS is perfomed as part of the SEC-MCR checklist for Attachment 1 of PEP-230, but this is not one of the non-delegable responsibilities The second item listed is plausible because if the event classification is an Site Area Emergency the ERO facilities are required to perform accountability within 30, but this is not one of the non-delegable responsibilities.

G2.4.40 Knowledge of SRO responsibilities in emergency plan implementation.
(CFR: 41.10 / 43.5 / 45.11)

Importance Rating:	2.7 4.5
Technical Reference:	PEP-230, Rev. 20, pg 26, Attachment 1
References to be provided:	None
Learning Objective:	EP-LP-3.0, Obj 1 (SRO Only)
Question Origin:	NEW
Comments:	None
Tier/Group:	T3
SRO Justification:	Shift Manager fills the role of the SEC-MCR until relieved by the SEC-TSC. SRO must be knowledgable in the responsiblities of the SEC-MCR during an event. This is an SRO Only objective for lesson plan EP-LP-3.0.

100. Given the following plant conditions:
- The plant is operating at 97% power
- AOP-001, Malfunction of Rod Control and Indication System is being implemented for a suspected misaligned rod
- Rod H2 is misaligned ~~from it's bank~~ by 6 steps
- OST-1039, Calculation of Quadrant Power Tilt Ratio, has been completed
- The QPTR has been determined to be 1.07

Which ONE of the following (1) describes the expected alarm for the above conditions, AND (2) the MINIMUM Thermal Power reduction that must be performed in the next 2 hours in accordance with Technical Specifications?

(Reference provided)

A. (1) ALB-013-5-3, Power Range Upper Detector High Flux DEV Or Auto Defeat

(2) Reduce Thermal Power to 76%

B. (1) ALB-013-5-3, Power Range Upper Detector High Flux DEV Or Auto Defeat

(2) Reduce Thermal Power to 79%

C. (1) ALB-013-8-5, Computer Alarm Rod DEV/SEQ NIS PWR Range Tilts

(2) Reduce Thermal Power to 76%

D. (1) ALB-013-8-5, Computer Alarm Rod DEV/SEQ NIS PWR Range Tilts

(2) Reduce Thermal Power to 79%

Reason answer is correct: ALB-013-5-3, Power Range Upper Detector High Flux DEV Or Auto Defeat takes input from the Nuclear Instruments and calculates the Quadrant Power Tilt Ratio (QPTR) and alarms when the calculated value exceeds its required limit. The APP responds for ALB-013-5-3 directs the operator to reference Technical Specification 3.2.4. With the unit in Mode 1, above 50% of Rated Thermal Power (RTP) and QPTR above 1.02 but less that or equal to 1.09 the unit thermal power must be reduced from RTP 3% for each 1% the QPTR is in excess of 1. For these conditions the require power is 79% of RTP [100 - 21 (7% above 1 x 3%) = 79]

A *Incorrect. The first part is correct. The second part is plausible because the calculation for the final pwer level is based on subtracting the required value of reduction from the current reactor power vice the RTP, i.e. 100% power.*

B *Correct.*

C *Incorrect. The first part is pausible because the AFD monitor is an ERFIS generated computer alarm based on the input from the Nuclear Instruments similar to the QPTR monitor. The second part is plausible because the calculation for the final pwer level is based on subtracting the required value of reduction from the current reactor power vice the RTP, i.e. 100% power.*

D *Incorrect. The first part is pausible because the AFD monitor is and ERFIS generated computer alarm based on the input from the Nuclear Instruments similar to the QPTR monitor. The second part is correct.*

G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.
(CFR: 41.10 / 43.5 / 45.3 / 45.12)

Importance Rating:	4.2 4.2
Technical Reference:	APP-ALB-013, Rev. 33, pg 22, 23, 36, and 37 Technical Specification 3.2.4, pg 3/4 2-11 to 3/4 2-13 (pg 132 -134)
References to be provided:	None
Learning Objective:	NIS Student Text, Obj 12
Question Origin:	Bank
Comments:	None
Tier/Group:	T3
SRO Justification:	Requires knowledge of a Technical Specification action that is greater than 1 hour