1 POINT

Question 76

Unit 2 initial conditions:

- Time = 1300
- Reactor power = 100%

Current conditions:

- Time = 1310
- RCS pressure = 88 psig slowly decreasing
- Reactor Building pressure = 35 psig slowly increasing
- All SCM's = 0°F
- 2RIA-57 = 1.9 R/hr slowly increasing
- 2RIA-58 = 1.0 R/hr slowly increasing
- Average of 5 highest CETC's = 430˚F decreasing
- All RBCU's cannot be started
- All RBS Pumps cannot be started

Based on the above conditions, which ONE of the following is the Emergency Action Level Classification?

REFERENCE PROVIDED

A. Notification of Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Question 76
T1/G1 - cpw

011EG2.4.41, Large Break LOCA
Knowledge of the emergency action level thresholds and classifications.
(2.9/4.6)

K/A MATCH ANALYSIS
Requires classifying event based on EAL criteria using RP/1000/001

SRO-ONLY ANALYSIS
Requires assessing plant conditions and applying those conditions to various enclosures in RP/1000/001 to determine the correct Emergency Action Level (which is an SRO only function)

ANSWER CHOICE ANALYSIS
Answer: B

A. Incorrect: Plausible since this would be the correct answer based on Encl. 4.2 System Malfunctions.

B. **CORRECT: Per RP/0/B/1000/001 Att. 4.1 – 5 points for loss of RCS barrier using either RIA readings OR LOSCM. Another 1 point from containment due to no RBCU or RBS. Total of 6 points which is an ALERT per matrix at bottom of Encl. 4.1**

C. Incorrect: Plausible since this would be the correct answer if you added the 4 points and 5 points allowed per each column of RCS Barriers then added the 1 point from containment.

D. Incorrect: Plausible since this would be the correct answer if you took both 5 point blocks from RCS barrier and then added the 1 point from containment.

Technical Reference(s): **RP/0/B/1000/001**

Proposed references to be provided to applicants during examination:
RP/0/B/1000/001 Attachments 4.1 – 4.8

Learning Objective: **EAP-SEP R12**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 77

Unit 1 plant conditions:

- Reactor power = 100%
- CC CRD Return Flow = 125 gpm slowly decreasing
- CC Return Flow = 563 gpm decreasing
- Letdown Temperature = 138°F stable
- The following alarms have actuated:
  - 1SA-9/B-1 (CC CRD Return Flow Low).
  - 1SA-2/C-1 (HP Letdown Temperature High)
  - 1SA-9/C-1 (CC Component Cooling Return Flow Low)

Based on the above conditions:

(1) Which ONE of the following describes a required action from the associated ARG(s) to mitigate this event?

AND

(2) If the ARG directed actions are NOT successful, which Abnormal Procedure is required to be entered <u>first</u>?

A. (1) Decrease letdown flow using 1HP-7 (LETDOWN CONTROL)
(2) AP/32 (Loss of Letdown)

B. (1) Decrease letdown flow using 1HP-7 (LETDOWN CONTROL)
(2) AP/20 (Loss of Component Cooling)

C. (1) Verify CC Surge Tank level > 12 inches and start the standby CC pump
(2) AP/32 (Loss of Letdown)

D. (1) Verify CC Surge Tank level > 12 inches and start the standby CC pump
(2) AP/20 (Loss of Component Cooling)

Question 77

T1/G1 - cpw

026 2.1.23, Loss of Component Cooling Water **Ability to perform specific system and integrated plant procedures during all modes of plant operation.** (4.3/4.4)

K/A MATCH ANALYSIS

Requires the ability to perform ARG procedures as well as AP's that are applicable as a result of loss of Component Cooling water

SRO-ONLY ANALYSIS

Requires assessment of facility conditions and selection of appropriate procedures during an abnormal situation when multiple entry conditions are satisfied. (43.5)

ANSWER CHOICE ANALYSIS

Answer: D

1SA-2/C-1 setpoint = 130°F and is actuated. 1SA-9/B-1 (CC CRD Return Flow Low) setpoint = 138 gpm and is actuated. 1SA-9/C1 setpoint = 575 gpm and is actuated. Loss of letdown has occurred as a result of decreased CC flow. SRO must prioritize AP entry.

A. Incorrect: Plausible in that the action directed is correct for 1SA-2/C-1 (High Letdown Temp). Entry conditions are met for both AP/20 and AP/32 since letdown auto isolates at 135 degrees however; AP/20 is first to be entered by statalarm procedure (AP/20 is the higher priority procedure and must be prioritized first).

B. Incorrect: Plausible in that the action directed is correct for 1SA-2/C-1 (High Letdown Temp). Abnormal procedure selection is correct.

C. Incorrect: First part is correct. Action is correct as it is directed by 1SA-9/B1. Second part is incorrect. Entry conditions are met for both AP/20 and AP/32 however; AP/20 is first to be entered by statalarm procedure (AP/20 is the higher priority procedure and must be prioritized first).

D. **Correct: Action to verify CC surge tank level > 12 inches and start standby CC pump are directed in 1SA-9/B-1. AP/20 is first to be entered by statalarm procedure.**

Technical Reference(s): **OP/3/A/6103/002 (3SA-2/C-1), OP/3/A/6103/009 (3SA-9/B-1 & 3SA-9/C-1), AP/20, AP/32**

Proposed references to be provided to applicants during examination: **None**

Learning Objective: **PNS-CC Obj R17 & R20**

Question Source: **Bank**

Question History: Last NRC Exam **2009 Question 79 (Re-ordered answers)**

Question Cognitive Level: **Comprehension or Analysis**

1 POINT

Question 78

Unit 1 initial conditions:

- Date = 12/01
- Time = 0800
- Reactor Power = 100%
- All Pressurizer heater breakers trip open and cannot be reclosed

Current conditions:

- Date = 12/04
- Time = 2200

Based on the above conditions, which ONE of the following describes the MAXIMUM RCS temperature (degrees F) that is in compliance with the requirements of Tech Specs AND the bases for the Pzr heater power supply requirements in TS 3.4.9 (Pressurizer)?

REFERENCE PROVIDED

A. 320 / ensure ability to perform a subcooled cooldown while on natural circulation during an extended loss of power

B. 320 / ensure ability to control pressure and remain subcooled during an extended loss of power

C. 375 / ensure ability to perform a subcooled cooldown while on natural circulation during an extended loss of power

D. 375 / ensure ability to control pressure and remain subcooled during an extended loss of power

Question 78

T1/G1 - cpw

027AA2.09, Pressurizer Pressure Control System (PZR PCS) Malfunction **Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions:** Reactor power (3.5/3.6)

K/A MATCH ANALYSIS

Requires knowledge of the impact of a malfunction of the pressurizer heaters on reactor power. Since there is no direct correlation between a failure of pressurizer heaters on Reactor Power, the mandated TS requirement to shutdown based on the failure and the ability to determine the required reactor power is used here.

SRO-ONLY ANALYSIS

Requires knowledge from TS 3.4.9 Bases regarding the reason Pressurizer heaters are required to be able to be powered from emergency power supply

ANSWER CHOICE ANALYSIS

Answer: D

A. Incorrect: First part is plausible from two different error paths. Either using the shutdown requirement based on pressurizer level not meeting the TS or misapplying Condition D and not adding the additional 6 hrs once in MODE 3 to get below 325 degrees would lead to this choice. Second part is plausible since maintaining SCM during a natural circulation cooldown is a significant concern and having adequate pressurizer heaters is an integral part of being able to do that. Additionally plausible since it is generally always preferred to get the unit cooled down and to LPI.

B. Incorrect: First part is plausible from two different error paths. Either using the shutdown requirement based on pressurizer level not meeting the TS or misapplying Condition D and not adding the additional 6 hrs once in MODE 3 to get below 325 degrees would lead to this choice. Second part is correct.

C. Incorrect: First part is plausible from two different error paths. Either using the shutdown requirement based on pressurizer level not meeting the TS or misapplying Condition D and not adding the additional 6 hrs once in MODE 3 to get below 325 degrees would lead to this choice.. Second part is plausible since maintaining SCM during a natural circulation cooldown is a significant concern and having adequate pressurizer heaters is an integral part of being able to do that. Additionally plausible since it is generally always preferred to get the unit cooled down and to LPI.

D. **CORRECT: TS 3.4.9 Condition C allows 72 hours to restore pzr heater capacity. Then Condition D allows 12 hours to MODE 3. Since the elapsed time in the question is 86 hours, being in MODE 3 but above 325 degrees (since an additional 6 hours is allowed to get below 325 degrees) meets the requirement of the TS. TS 3.4.9 Bases says the requirement for emergency power supplies is based on NUREG-0737. The intent is to allow maintaining the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power.**

Technical Reference(s): **ADM-TSS, Tech Spec 3.4.9 and its bases**

Proposed references to be provided to applicants during examination: **TS 3.4.9 – Pressurizer**

Learning Objective: **ADM-TSS R1, R5**

Question Source: **NEW**

Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 79

Unit 1 initial conditions:

- Reactor power = 75% stable
- The following alarms actuate:
  - 1SA-8/A3 (FWPT A TRIP)
  - 1SA-8/A6 (FWPT B TRIP)
  - 1SA-1/A1 (RP CHANNEL A TRIP)
  - 1SA-1/B1 (RP CHANNEL B TRIP)
  - 1SA-1/C1 (RP CHANNEL C TRIP)
  - 1SA-1/D1 (RP CHANNEL D TRIP)

Current conditions:

- Reactor power = 7% slowly decreasing
- All SCM's = 0°F stable

Based on the above conditions, which ONE of the following describes the EOP tab that will be directed <u>first</u> AND subsequent conditions that would require a transfer to a different EOP tab?

A. UNPP / Reactor Vessel head level = 0 inches

B. UNPP / Station Blackout

C. LOSCM / Reactor Vessel head level = 0 inches

D. LOSCM / Station Blackout

Question 79
T1/G1 - cpw

029EA2.02, ATWS **Ability to determine or interpret the following as they apply to an ATWS:** Reactor trip alarm (4.2/4.4) 43.5

K/A MATCH ANALYSIS
Requires interpreting Rx trip alarms (RPS channel trips) and indications (Rx power) to determine if an ATWS is in progress.

SRO-ONLY ANALYSIS
Requires assessment of plant conditions and prescribing a procedure section with which to proceed.

ANSWER CHOICE ANALYSIS
Answer: B

A. Incorrect: First part is correct. Second part is plausible since head level of 0 inches is an indication of inadequate core cooling conditions. The ICC tab is a very high priority tab however it is not a higher priority than the UNPP tab therefore there is no transfer to ICC from UNPP.

B. CORRECT: Since an ATWS has occurred, Rule 1 will be entered and Rule 1 will direct the PD to go to the UNPP tab. Even if Rule 1 is missed the PA page of SA will direct the SRO to UNPP. A blackout and therefore transfer to the Blackout tab is the only transfer out of UNPP allowed.

C. Incorrect: First part is plausible since the IC's provide the operator with a loss of subcooling margin condition. LOSCM tab is a high priority tab however the UNPP tab takes precedence therefore UNPP would be the correct tab.

D. Incorrect: Plausible since the entry conditions for the LOSCM tab are met however UNPP tab is a higher priority both from Rule 1 and the PA page of SA.

Technical Reference(s): **EOP-SA EOP-UNPP, Rule 1**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-UNPP R1, R10, & R12**

Question Source: **New**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 80

Unit 1 plant conditions:

- Reactor power = 100%
- 1DCA and 1DCB supply breakers to 1KI inverter tripped open
- 1KI inverter static transfer switch fails to transfer power
- AP/23 (Loss Of ICS Power) in progress

Based on the above conditions, which ONE of the following describes conditions where AP/23 would direct tripping all operating Main Feedwater pumps AND the Tech Spec bases for the "Loss Of Main Feedwater Pumps" RPS anticipatory trip?

Both MFDW pumps are required to be tripped if _____ AND the Tech Spec bases for the "Loss Of Main Feedwater Pumps" RPS anticipatory trip is to reduce the possibility of _____.

A. Steam Generator pressure is unstable / challenging the PORV

B. Steam Generator pressure is unstable / operation at power while on Emergency Feedwater

C. the reactor trips / challenging the PORV

D. the reactor trips / operation at power while on Emergency Feedwater

Question 80

T1/G1 - cpw

058G2.4.11, Loss of DC Power **Knowledge of abnormal condition procedures.** (4.0/4.2)

K/A MATCH ANALYSIS

Requires knowledge of mitigation strategy in AP/23 when a loss of DC supply to KI inverter occurs.

SRO-ONLY ANALYSIS

Requires detailed knowledge of decision path regarding when to trip MFDWP's and this action would require transition to the event specific Rule 3 due to loss of main feedwater. Also requires knowledge from the bases of Tech Specs regarding the purpose of the Loss of Main Feedwater Pump trip. Although both answers could be correct from a systems knowledge perspective, it is the determination of the Bases for the trip that makes this SRO level.

ANSWER CHOICE ANALYSIS

Answer: C

A. Incorrect: First part is plausible since this condition is addressed in the applicable section of AP/23 (Loss of ICS Power) only corrective actions requires adjusting Turbine Master rather than tripping MFDWP's. Second part is correct

B. Incorrect: First part is plausible since this condition is addressed in the applicable section of AP/23 (Loss of ICS Power) only corrective actions requires adjusting Turbine Master rather than tripping MFDWP's. Second part is plausible since EFDW cannot provide sufficient flow to support power operations. Additional plausibility comes from the fact that there is a required manual reactor trip if both MFDWP's trip and CR's are > 50% Gp1. The consequences of being at power and the Rx not tripping when Main FDWP's trip would be EFDW providing water to SG's at power therefore it would be plausible to deduce that the reason the Rx is tripped when both MFDWP's trip is to prevent operation when EFDW was providing flow to SG's.

C. **CORRECT: AP/23 section 4B for Loss of ICS Auto Power directs via an IAAT that if Rx trips then trip both MFWP's. The bases for TS 3.3.1 (Reactor Protective System Instrumentation) describes the purpose of the trip as being "to minimize challenges to the PORV"**

D. Incorrect: First part is correct. Second part is plausible since EFDW cannot provide sufficient flow to support power operations. Additional plausibility comes from the fact that there is a required manual reactor trip if both MFDWP's trip and CR's are > 50% Gp1. The consequences of being at power and the Rx not tripping when Main FDWP's trip would be EFDW providing water to SG's at power therefore it would be plausible to deduce that the reason the Rx is tripped when both MFDWP's trip is to prevent operation when EFDW was providing flow to SG's.

Technical Reference(s): **AP/23 Loss of ICS Power, Tech Spec 3.3.1**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-APG R8, ADM-TSS R5**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 81

Unit 1 initial conditions:

- Reactor trip from 100% power
- Both Main Feedwater pumps trip
- EFDW pumps will not start

Current conditions:

- Rule 3 in progress
- LOHT tab initiated
- ALL SCM's > 0°F

Based on the above conditions, which ONE of the following changes in plant conditions would require an immediate transfer out of the LOHT tab?

A. TDEFWP becomes available

B. Condensate Booster Pump feed established

C. Indications of a Steam Generator Tube Rupture in the 1A SG

D. Core SCM = 0°F due to RCS heatup AND HPI Forced Cooling established

Question 81
T1/G1 - cpw

BE04EA2.1, Inadequate Heat Transfer-Loss of Secondary Heat Sink
Ability to determine or interpret the following as they apply to the (Inadequate Heat Transfer):
Facility conditions and selection of appropriate procedures during abnormal and emergency operations
(3.0/4.2)

K/A MATCH ANALYSIS
Must determine plant conditions and select correct procedure path required based on those conditions.

SRO-ONLY ANALYSIS
Requires assessing plant conditions and prescribing a section of the procedure with which to proceed

ANSWER CHOICE ANALYSIS
Answer: D

A. Incorrect: Plausible since the availability of the TDEFWP can eliminate the loss of heat transfer condition and guidance for feeding the SG's with the TDEFWP with no other major issues can be found in the Subsequent Actions tab so it would be plausible to assume a transfer to SA would be appropriate however guidance for re-establishing feed to SG's is in the LOHT tab.

B. Incorrect: Plausible since once CBP feed is established there is no longer a loss of heat transfer and it would be plausible to deduce a transfer back to the SA tab would be appropriate however the guidance needed for CBP feed is in Rule 2 and the LOHT tab has you stay there during CBP feed until another source of FDW is available

C. Incorrect: Plausible since a transfer to the SGTR tab would be correct if it did not occur in conjunction with the LOHT. Normally, if a SGTR occurs and the EOP is entered based on those indications the SGTR tab would provide the mitigation strategy.

D. CORRECT: Transfer to LOSCM does not occur here if the reason Core SCM = 0 is due to RCS heatup as a result of the LOHT. When this occurs, LOHT directs performing Rule 4. As long as HPI operation is normal when Rule 4 is complete, LOHT directs a transfer to the HPI CD tab.

Technical Reference(s): **EOP LOHT tab page 1, EAP-LOHT ATTACH. 1 & 5**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **EAP-LOHT R22**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **COMPREHENSION AND ANALYSIS**

1 POINT

Question 82

Unit 1 initial conditions:

- Startup in progress
- Reactor in MODE 3
- Source Range NI's 2 & 4 OOS
- Source Range NI's 1 & 3 = 834 cps increasing
- Wide Range NI's 1 thru 4 = 5E-3% increasing

Current conditions:

- Source Range NI-1 = 0 cps
- 1SA-05/A8 (NI-1 TEST/FAIL) actuates

Based on the above conditions, which ONE of the following describes a reason for the NI-1 indications AND the required status of Steam Generator level control in accordance with Tech Spec 3.7.5 (Emergency Feedwater) bases?

A. Loss of +15V Power supply / ONLY manual level control required in MODE 3 & 4

B. Loss of +15V Power supply / BOTH automatic AND manual level control required in MODE 3 & 4

C. Source range becomes saturated / ONLY manual level control required in MODE 3 & 4

D. Source range becomes saturated / BOTH automatic AND manual level control required in MODE 3 & 4

Question 82
T1/G2 -cpw

032G2.4.45, Loss of Source Range NI
Ability to prioritize and interpret the significance of each annunciator or alarm.
(4.1/4.3)

K/A MATCH ANALYSIS
OK to ask about determining problems with NI even if no annunciator indicates the problem. Follow up with Gerry if needed.
Requires ability to interpret the significance of Source Range annunciators.

SRO-ONLY ANALYSIS
Requires knowledge of bases information from TS 3.7.5 (EFW)

ANSWER CHOICE ANALYSIS
Answer: A

A. **CORRECT: Per 1SA-5/A8 if any power supply is low, NI-1 fails low and the alarm is actuated. Automatic level control is required to be operable by TS 3.7.5 anytime the reactor is in MODE 2 or above however. Only manual level control is required in MODE 3 and 4.**

B. Incorrect: First part is correct. Second part is plausible since both automatic and manual level control are required however only manual control is required in MODE 3 and 4.

C. Incorrect: First part is plausible since Source Range NI's do become saturated during a Rx startup. Since some Nuclear Instrumentation resets to 0 cps when saturated to protect the instrument (as with previous versions of our SR NI's) it is reasonable to believe that they would indicate 0 cps when saturated. However with our current NI's the Dixon gages will just be pegged high. Second part is correct.

D. Incorrect: First part is plausible since Source Range NI's do become saturated during a Rx startup. Since some Nuclear Instrumentation resets to 0 cps when saturated to protect the instrument (as with previous versions of our SR NI's) it is reasonable to believe that they would indicate 0 cps when saturated. However with our current NI's the Dixon gages will just be pegged high. Second part is plausible since both automatic and manual level control is required however only manual control is required in MODE 3 and 4.

Technical Reference(s): **IC-NI, 1SA-5/A8, TS 3.7.5 bases**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **IC-NI R9, 10 ADM-ITS R3**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 83

Unit 1 plant conditions:

- RCS temperature = 544°F stable
- Safety Rod groups withdrawal in progress
- ALL Wide Range NI's fail LOW

Which ONE of the following describes the …

(1) ONE hour action required by TS 3.3.10 (Wide Range Neutron Flux)
AND
(2) power levels at which the Tech Spec bases credits the WR NI's to alert operators of reactivity transients in anticipation of RPS actuation?

A. reduce power to <4E-3% RTP / low power

B. reduce power to <4E-3% RTP / high power

C. Open CRD breakers / low power

D. Open CRD breakers / high power

Question 83
T1/G2 -cpw

033AG2.1.23, Loss of Intermediate Range NI **Ability to perform specific system and integrated plant procedures during all modes of plant operation.** (4.3/4.4) 43.5

K/A MATCH ANALYSIS
Requires performing actions based on administrative procedures (TS 3.3.10) as well as understanding what function TS credits the WR NI's with doing.

SRO-ONLY ANALYSIS
Requires knowledge of bases information for TS 3.3.10 Wide Range Neutron Flux

ANSWER CHOICE ANALYSIS
Answer: C

A. Incorrect: First part is plausible since the power reduction would be correct if only 1 required WR NI had failed (although 2 hours would be allowed to perform the action). Second part is correct

B. Incorrect : First part is plausible since the power reduction would be correct if only 1 required WR NI had failed (although 2 hours would be allowed to perform the action). Second part is plausible since at higher powers you are operating closer to RPS trip setpoints so it would be reasonable to deduce that additional measures to monitor for Reactivity transients would be required.

C. CORRECT: Per TS 3.3.10 (Wide Range Neutron Flux) CRD trip breakers must be opened within 1 hour. The bases of TS 3.3.10 describes the credited function as being alerting the operator to reactivity transients at low power to anticipate RPS actuation

D. Incorrect: First part is correct. Second part is plausible since at higher powers you are operating closer to RPS trip setpoints so it would be reasonable to deduce that additional measures to monitor for Reactivity transients would be required.

Technical Reference(s): **TS 3.3.10**
Proposed references to be provided to applicants during examination: NONE
Learning Objective: **ADM-TSS R4, R5**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 84

Unit 1 initial conditions:
- Reactor power = 100% stable

Current conditions:
- Reactor power = 28% decreasing
- Both Generator Output Breakers trip open

Which ONE of the following…

(1) procedure entry conditions are met based on current conditions?
AND
(2) states what the Atmospheric Dump Valves are credited for doing during a SGTR in accordance with Tech Spec bases?

A. AP/1 (Unit Runback) / Prevent exceeding Compressive stress limits on the isolated SG

B. AP/1 (Unit Runback) / Cooldown the RCS using BOTH SG's to allow isolating the ruptured SG within 40 minutes

C. EOP / Prevent exceeding Compressive stress limits on the isolated SG

D. EOP / Cooldown the RCS using BOTH SG's to allow isolating the ruptured SG within 40 minutes

Question 84

T1/G2 - cpw

BA01AA2.1, Plant Runback

Ability to determine or interpret the following as they apply to the (Plant Runback):

Facility conditions and selection of appropriate procedures during abnormal and emergency operations

(3.0/3.7)

K/A MATCH ANALYSIS

Requires ability to determine if a plant will occur and selecting the appropriate procedure based on that determination

SRO-ONLY ANALYSIS

Requires knowledge of the TS bases regarding how the ADV's are credited to help mitigate a SGTR.

ANSWER CHOICE ANALYSIS

Answer: B

A. Incorrect: First part is correct. Second part is plausible since compressive stresses during a SGTR scenario are valid concerns. Additional plausibility comes from the fact that tensile stresses in a SG are mitigated by steaming the SG and the ADV's are what are credited for providing a method of steaming the SG's since the TBV's are not safety related therefore the misconception of steaming is required to mitigate compressive stresses would lead to this choice.

B. **CORRECT: For power levels below 40%, the unit will survive a load rejection which means that the EOP entry conditions will not be met and therefore AP/1 (Unit Runback) will be the controlling procedure to stabilize the plant. The bases of TS 3.7.4 (ADV Flow Paths) explains that the ADV's are credited in the SGTR scenario to allow the operator to manually depressurize both SG's to allow isolation of the ruptured SG within 40 minutes of identifying the rupture.**

C. Incorrect: First part is plausible since for power levels > 40% a Rx trip is possible during a load rejection. For powers > 70% a Rx trip is considered imminent. Second part is plausible since compressive stresses during a SGTR scenario are valid concerns. Additional plausibility comes from the fact that tensile stresses in a SG are mitigated by steaming the SG and the ADV's are what are credited for providing a method of steaming the SG's since the TBV's are not safety related therefore the misconception of steaming is required to mitigate compressive stresses would lead to this choice.

D. Incorrect: First part is plausible since for power levels > 40% a Rx trip is possible during a load rejection. For powers > 70% a Rx trip is considered imminent. Second part is correct.

Technical Reference(s): **AP/1, TS 3.7.4 (ADV's)**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **STG-ICS R9, ADM-TSS R5**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 85

Unit 2 initial conditions:

- Reactor power = 100% stable
- Main Turbine trips
- Group 4 rod 6 remains 100% withdrawn

Based on the above conditions, which ONE of the following describes actions required by the EOP and the Tech Spec design bases for the Turbine Stop Valves?

A. Open 2HP-24 and 1HP-25 ONLY / Prevent overcooling during a MSLB

B. Open 2HP-24 and 1HP-25 ONLY / Isolate SG's during a SGTR

C. Initiate Emergency Boration / Prevent overcooling during a MSLB

D. Initiate Emergency Boration / Isolate SG's during a SGTR

Question 85
T1/G2 - cpw

BA04AA2.2, Turbine Trip
Ability to determine or interpret the following as they apply to the (Turbine Trip):
Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments
(3.7/3.7)

K/A MATCH ANALYSIS
Requires ability to adhere to steps in EOP-SA as a result of stuck rod during a Turbine Trip from power.

SRO-ONLY ANALYSIS
Requires knowledge from TS bases for Turbine Stop Valves.

ANSWER CHOICE ANALYSIS
Answer: A

A. **CORRECT: The turbine trip would require entry into EOP SA tab. The SA tab checks to ensure all control rods are inserted and if not directs opening 1HP-24 and 1HP-25.The bases of TS 3.7.2 (Turbine Stop Valves) describes the purpose of the TSV's as to stop steam flow to the turbine to prevent overcooling.**

B. Incorrect: First part is correct. Second part is plausible since separating the SG's is part of what the TSV's are credited for doing however it is in the context of limiting SG blowdown on a MSLB to prevent overcooling.

C. Incorrect: First part is plausible since initiating Emergency Boration is an action directed by the EOP but it is directed during an ATWS. With only 1 control rod stuck out and no indication of being at power, Emergency Boration would not be required. Second part is correct.

D. Incorrect: First part is plausible since initiating Emergency Boration is an action directed by the EOP but it is directed during an ATWS. With only 1 control rod stuck out and no indication of being at power, Emergency Boration would not be required. Second part is plausible since separating the SG's is part of what the TSV's are credited for doing however it is in the context of limiting SG blowdown on a MSLB to prevent overcooling.

Technical Reference(s): **ADM-TSS, TS 3.7.2 (Turbine Stop Valves), EAP-SA**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **ADM-TSS R5 and EAP-SA R1**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **Analysis and Comprehension**

1 POINT

Question 86

Unit 1 initial conditions:

- Reactor in MODE 3
- RCS leakage = 95 gpm
- 1A2 RCP Motor Upper Guide Bearing temperature = 195°F slowly increasing
- 1B2 RCP Motor Stator Temperature = 235°F slowly increasing

Based on the above conditions, which ONE of the following describes the actions required by AP/16 (Abnormal Reactor Coolant Pump Operation) AND which EOP tab would provide guidance if the 2nd RCP was secured and a Natural Circulation cooldown was required?

Trip the …

A. 1A2 RCP / Forced Cooldown

B. 1A2 RCP / LOCA Cooldown

C. 1B2 RCP / Forced Cooldown

D. 1B2 RCP / LOCA Cooldown

Question 86
T2/G1 - cpw

003A2.02, Reactor Coolant Pump
Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP
(3.7/3.9)

K/A MATCH ANALYSIS
Requires predicting the impact of the malfunctions on RCP operation and using AP/16 (Abnormal RCP Ops) to determine actions required to shutdown the RCP under the abnormal conditions

SRO-ONLY ANALYSIS
Requires assessing plant conditions (the need to secure RCP bases on ITC) and then prescribing a section of a procedure with which to proceed.

ANSWER: A

A. **CORRECT: The Immediate Trip Criteria (ITC) limit for the upper guide bearing temp is 190 degrees therefore the ITC has been exceeded. Per AP/16 if Rx power is <70% you trip the affected RCP. If the 2nd RCP is lost and therefore no RCP's are operating you will meet the entry conditions for the EOP. At the end of Subsequent Actions the procedure will determine if a Natural Circulation cooldown is required and if so direct you to the Forced Cooldown Tab to continue with the cooldown.**

B. Incorrect: First part is correct. Second part is plausible due to the large RCS leak being present however the criteria for using the LOCA Cooldown tab is to have a leak that is larger than normal RCS makeup which is approximately 160 gpm. Since this leak is not that large the LOCA CD tab is not utilized.

C. Incorrect: The temp limit for motor stator is 295 degrees. 235 degrees is plausible since there are several ITC trip setpoints with temps less than 235 degrees (examples are Radial Bearing temps (225) and thrust bearing temps (190). If you had the misconception that the ITC for the 1B2 were met then this would be correct.

D. Incorrect: The temp limit for motor stator is 295 degrees. 235 degrees is plausible since there are several ITC trip setpoints with temps less than 235 degrees (examples are Radial Bearing temps (225) and thrust bearing temps (190). Second part is plausible due to the large RCS leak being present however the criteria for using the LOCA Cooldown tab is to have a leak that is larger than normal RCS makeup which is approximately 160 gpm. Since this leak is not that large the LOCA CD tab is not utilized.

Technical Reference(s): **AP/16 (Abnormal RCP Operation) AP/29 Rapid Unit Shutdown**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **EAP APG R8, R9**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 87

Unit 3 plant conditions:

- Reactor power = 100%
- 3A RBCU tagged out
- 3LPSW-19 (3B RBCU INLET) fails closed

Based on the above conditions, which ONE of the following describes the capabilities of the Reactor Building Spray and Cooling systems to perform their credited safety function during a LOCA and the actions required to comply with Tech Spec 3.6.5 (Reactor Building Spray and Cooling Systems)?

Following a LOCA, Reactor Building Spray and Cooling systems _____ perform their credited safety functions AND _____.

A. can / Restore Containment to OPERABLE status within 1 hour

B. can / Immediately enter LCO 3.0.3

C. cannot / Restore Containment to OPERABLE status within 1 hour

D. cannot / Immediately enter LCO 3.0.3

Question 87
T2/G1 -cpw

022A2.04, Containment Cooling **Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:** Loss of service water (2.9/3.2)

K/A MATCH ANALYSIS
Requires predicting the impact that losing LPSW to two of the RBCU's will have on Containment Cooling's ability to perform its safety function and then requires choosing the correct procedures to mitigate the consequences of the failure.

SRO-ONLY ANALYSIS
Requires knowledge from the bases of TS 3.6.5 to determine if sufficient RB Cooling systems are available to perform their credited safety function. Also requires knowledge of the requirements of LCO 3.0.3 since TS 3.6.5 Condition H requires immediate entry into LCO 3.0.3.

ANSWER CHOICE ANALYSIS
Answer: D

A. **Incorrect:** First part is plausible since there are still two RBS trains and one RBCU train operable. If you assumed that RBS and RBCU's are redundant systems then you would deduce that safety function is still available since both RBS trains remain operable. Second part is plausible since the malfunction would result in a RBCU Cooler Rupture alarm since the LPSW-21 (outlet valve) is still open. The ARG for the alarm will direct assessing indication to see if a cooler rupture has occurred. This choice could be correct if there were indications of an actual cooler rupture and the cooler is not correctly isolated.
B. **Incorrect:** First part is plausible since there are still two RBS trains and one RBCU train operable. If you assumed that RBS and RBCU's are redundant systems then you would deduce that safety function is still available since both RBS trains remain operable. Second part is correct. It also has plausibility if you assume the first part is correct since not all TS required entries into LCO 3.0.3 occur as a result of a loss of safety function of the subject SSC.
C. **Incorrect:** First part is correct. Second part is plausible per explanation in A.
D. **CORRECT: Per bases of TS 3.6.5 the safety analysis assumes at least one RBS train and 2 RBCU are available following a LOCA therefore the minimum required to meet the safety function is not available. TS 3.6.5 Condition H directs immediate entry into LCO 3.0.3.**

Technical Reference(s): **TS 3.6.5 and its bases, ARG fir 1SA9/C9**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **ADM-TSS R5, R6**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 88

Unit 3 initial conditions:

- Time = 1200
- Reactor power = 100%

Current conditions:

- Time = 1201
- RCS Pressure = 86 psig slowly decreasing
- RB Pressure = 28 psig slowly increasing
- 3B Reactor Building Spray (RBS) pump failed to start

Based on the above conditions, which ONE of the following predicts the impact of the 3B RBS pump failure on Iodine concentration in containment following a LOCA and describes actions required of the RO by EOP Enclosure 5.1 (ES Actuation)?

Iodine concentrations assumed in the Safety Analysis for a Large Break LOCA _____ be exceeded and EOP Enclosure 5.1 _____ require SRO approval to start the 3B RBS pump.

A. will / does

B. will / does NOT

C. will NOT / does

D. will NOT / does NOT

Question 88

T2/G1 - cpw

026A2.04, Containment Spray System
Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Failure of spray pump
(3.9/4.2)

K/A MATCH ANALYSIS

Requires predicting the impact of the failure of a RBS pump on the consequences of a LBLOCA and knowledge of procedure mitigation of the failure. In the case of Encl. 5.1 it is more than just OK not to take actions on components that do not reposition, it is a required action not to take the actions since it would impede progress through the enclosure and jeopardize meeting TCA's.

SRO-ONLY ANALYSIS

Requires knowledge from TS bases regarding minimum RBS Pump requirements to perform Safety Function assumed for RBS

ANSWER CHOICE ANALYSIS

Answer: C

A. Incorrect:First part is plausible since ~1/2 RBS flow has been lost and not all systems at Oconee that are credited in safety analysis have a redundant train that can be lost and still accomplish the safety function in question. Second part is correct.
B. Incorrect: Both parts are incorrect. First part is plausible per A. Second part is plausible since some failed ES components (example LPIP's) are addressed directly in Encl 5.1 and do not require SRO approval to restart/reposition.
C. **One reactor building spray train is required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. (ADM-TSS page 94) EOP Encl 5.1 directs the RO to notify SRO to evaluate starting components not in their ES position and does NOT direct the RO to attempt to start the pump. It is critical to the success of Encl. 5.1 that the RO does not take the time to attempt to start the RBS pump to ensure TCA's in Encl 5.1 are met. Based on this requirement, it is actually a required action to NOT take the time to start the RBS pump and SRO approval would be required for the RO to start the pump since there is no direct procedural guidance.**
D. Incorrect: First part is correct. Second part is plausible since some failed ES components (example LPIP's) are addressed directly in Encl 5.1. It is critical to the success of Encl. 5.1 that the RO does not take the time to attempt to start the RBS pump to ensure TCA's in Encl 5.1 are met. Based on this requirement, it is actually a required action to NOT take the time to start the RBS pump. Second part is plausible since some failed ES components (example LPIP's) are addressed directly in Encl 5.1 and do not require SRO approval to restart/reposition.

Technical Reference(s): **TS 3.6.5 bases, ADM-TSS, EOP Encl. 5.1**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **ADM-TSS R5, EAP-ESA R9**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **Analysis and Comprehension**

1 POINT

Question 89

Unit 1 plant conditions:

- Reactor power = 20% stable
- 1A Main Feedwater Pump in service

Based on the above conditions, which ONE of the following describes conditions that will require entry into the EOP AND the initial actions required by the EOP if one MSRV does NOT reseat.

A. FDWPT Exhaust Vacuum = 23.2 inches HG / Transfer to EHT tab

B. FDWPT Exhaust Vacuum = 23.2 inches HG / Manually decrease SG pressure

C. Feedwater pump bearing oil pressure = 3.1 psig / Transfer to EHT tab

D. Feedwater pump bearing oil pressure = 3.1 psig / Manually decrease SG pressure

Question 89
T2/G1 - cpw

059G2.2.42, Main Feedwater
Knowledge of system set points, interlocks, and automatic actions associated with EOP entry conditions.
(4.5/4.6)

K/A MATCH ANALYSIS
Requires knowledge of Main Feedwater pump trip setpoints and knowledge of EOP entry conditions.

SRO-ONLY ANALYSIS
Requires assessing abnormal plant conditions and prescribing a section of a procedure with which to proceed. Internal steps in the Subsequent Actions tab will have the SRO determine if a MSRV is stuck open and if so will provide guidance to reduce MS pressure in steps to reseat the valve. The actions are NOT part of entry conditions and are more detailed than major mitigation strategy.

ANSWER CHOICE ANALYSIS
Answer: D

A. Incorrect: First part is plausible since the lo vacuum alarm setpoint is 25 inches. The misconception that the lo vacuum trip for either the main turbine (21.75") or MFDWP turbine (19") is 25 inches would lead to this choice. Second part is plausible since a stuck open MSRV could exhibit characteristics of a main steam line leak/break and without specific guidance in the EOP could be determined to be EHT therefore a transfer to the EHT tab would be plausible.

B. Incorrect: First part is plausible since the lo vacuum alarm setpoint is 25 inches. The misconception that the lo vacuum trip for either the main turbine (21.75") or MFDWP turbine (19") is 25 inches would lead to this choice. Second part is correct.

C. Incorrect: First part is correct. Second part is plausible since a stuck open MSRV could exhibit characteristics of a main steam line leak/break and without specific guidance in the EOP could be determined to be EHT therefore a transfer to the EHT tab would be plausible.

D. **CORRECT: The low bearing oil pressure trip for a main feedwater pump is ≤ 4 psig therefore the operating FDW pump would trip. With no Feedwater pumps operating a Rx trip would occur requiring entry into the EOP. Once in SA's MSRV status is verified. If any MSRV is still open, directions are given to reduce SG pressure in 10 psi increments in an attempt to reseat the valve with a limit on decreasing pressure being maintain RCS temp ≥ 532 degrees.**

Technical Reference(s): **CF-FPT, STG-EHC, EOP**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **CF-FDW R8,STG-EHC R10, EAP-SA R15**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **knowledge and fundamentals**

1 POINT

Question 90

Unit 1 initial conditions:

- Unit shutdown in progress
- RCS pressure = 1550 slowly decreasing
- RCS temperature = 495 degrees F slowly decreasing

Current conditions:

- LBLOCA occurs
- RCS pressure = 20 psig slowly decreasing
- RB pressure = 3.4 psig slowly increasing
- SCM's = 0°F stable

Based on the above conditions, which ONE of the following describes whether Containment isolation is complete AND if HPI is required by TS bases to mitigate the event?

ASSUME NO OPERATOR ACTIONS

Containment isolation _____ complete AND HPI _____ required to mitigate the LBLOCA.

A. is / is

B. is / is NOT

C. is NOT / is

D. is NOT / is NOT

Question 90
T2/G1 - cpw

103G2.4.9, Containment **Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.** (3.8/4.2)

K/A MATCH ANALYSIS
Requires knowledge of the impact of being below 1600 psig when a LOCA occurs on containment isolation
SRO-ONLY ANALYSIS
Requires knowledge of the design bases of HPI injection as described in the bases of TS 3.5.2 (HPI).
ANSWER CHOICE ANALYSIS
Answer: B

A. Incorrect: First part is correct. Second part is plausible since it would be correct if the question were asking about a SBLOCA however our LBLOCA analysis does not take any credit for HPI injection since the RCS would depressurize to allow LPI to inject.

B. **CORRECT: Although ES HPI would already be bypassed by the shutdown procedure (done between 1675 psig and 1600 psig), only the RCS pressure portion of ES 1&2 is bypassed. With ES HPI bypassed, RB pressure reaching 3 psig would still result in an ES 1-6 actuation and therefore a complete RB isolation. The bases of TS 3.5.2 (HPI) explains that while HPI is credited to mitigate a SBLOCA, it is not credited in the analysis for a LBLOCA.**

C. Incorrect: First part is plausible since ES HPI bypass occurs during a unit shutdown between 1675 and to 1600 psig however only the RCS pressure input to ES 1&2 are bypassed. Since RB pressure is > 3 psig, ES 1&2 would still actuate. The misconception that the HPI bypass function will prevent RB pressure from actuating ES 1&2 would result in this choice. Second part is plausible since it would be correct if the question were asking about a SBLOCA however our LBLOCA analysis does not take any credit for HPI injection since the RCS would depressurize to allow LPI to inject.

D. Incorrect: First part is plausible since ES HPI bypass occurs during a unit shutdown between 1675 psig and 1600 psig however only the RCS pressure input to ES 1&2 are bypassed. Since RB pressure is > 3 psig, ES 1&2 would still actuate. The misconception that the HPI bypass function will prevent RB pressure from actuating ES 1&2 would result in this choice. Second part is correct.

Technical Reference(s): **TS 3.5.2 (HPI) IC-ES OP/1/A/1102/010 (Unit Shutdown Procedure**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **IC-ES R10 ADM-TSS R5**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **Comprehension and Analysis**

Question 91

Unit 1 plant conditions:

- Reactor power = 100%
- 1RC-1 fails open
- 1RC-3 will NOT close

Based on the above conditions, which ONE of the following describes the initial response of 1HP-120 (RC VOLUME CONTROL) valve position AND the action(s) required by AP/44 (Abnormal Pressurizer Pressure Control)?

1HP-120 valve position will....

A. remain approximately the same / Trip the reactor and secure the 1A1 and 1A2 RCP's

B. remain approximately the same / Throttle 1HP-120 and 1HP-7 as required to stabilize RCS pressure

C. throttle fully open / Trip the reactor and secure the 1A1 and 1A2 RCP's

D. throttle fully open / Throttle 1HP-120 and 1HP-7 as required to stabilize RCS pressure

Question 91
T2/G2 -cpw

011A2.06, Pressurizer Level Control
Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Inadvertent PZR spray actuation
(3.7/3.9)

K/A MATCH ANALYSIS
Requires predicting the impact that the failure of RC-1 will have on 1HP-120 and the procedural actions required to mitigate the consequences of the malfunctions.

SRO-ONLY ANALYSIS
Requires assessing plant conditions and prescribing a section of a procedure (path in AP/44 with RCS pressure continuing to decrease) to mitigate the event. Requires more than just major mitigation strategy.

ANSWER CHOICE ANALYSIS
Answer: A

A. **CORRECT: 1RC-3 failed open will have no impact on the operation of Pzr LCS since a 1RC-3 failure has no impact on the setpoint of 1HP-120, RCS temp, or the volume of water in the RCS. AP/44 directs manually tripping the reactor and once power is <1% the A1 and A2 RCP's are stopped to decrease spray flow.**

B. Incorrect: First part is correct. Second part is plausible since the actions described are used to stabilize RCS pressure however it is under different circumstances. If the pressurizer is subcooled, these actions are taken to stabilize pressure while the pressurizer is heated to saturation.

C. Incorrect: First part is plausible since it would be correct if the failure resulted in an immediate Rx trip. The failure would cause a Rx trip however since Pzr heaters are available, there is a significant delay before the RCS low pressure trip setpoint is reached. Second part is correct

D. Incorrect: First part is plausible since it would be correct if the failure resulted in an immediate Rx trip. The failure would cause a Rx trip however since Pzr heaters are available, there is a significant delay before the RCS low pressure trip setpoint is reached. Second part is plausible since the actions described are used to stabilize RCS pressure however it is under different circumstances. If the pressurizer is subcooled, these actions are taken to stabilize pressure while the pressurizer is heated to saturation

---

Technical Reference(s): **AP/44 and EOP**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **EAP-APG R9**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **Knowledge and Fundamentals**

1 POINT

Question 92

Unit 3 initial conditions:

- Reactor power = 100%
- The volume of active TSP required by SLC 16.6.10 (Trisodium Phosphate) is NOT met

Current conditions:

- LOCA/LOOP occurs

Based on the above conditions, which ONE of the following describes the consequence of the inadequate TSP volume AND when TSP performs its credited function?

A. EOP will direct going to single Reactor Building Spray pump operation / During the injection phase

B. EOP will direct going to single Reactor Building Spray pump operation / During the recirculation phase

C. Acceptable offsite dose limits may be exceeded / During the injection phase

D. Acceptable offsite dose limits may be exceeded / During the recirculation phase

Question 92
T2/G2 -cpw

027G2.2.40, Containment Iodine Removal **Ability to apply Technical Specifications for a system.** (3.4/4.7)

K/A MATCH ANALYSIS
Per NRC, ask about TSP SLC (16.6.10) bases. Requires applying information from the spec and bases of SLC 16.6.10 (TSP) to a specific situation.

SRO-ONLY ANALYSIS
Requires knowledge of bases information regarding TSP baskets for iodine removal per SLC 16.6 10

ANSWER CHOICE ANALYSIS
Answer: D

A. Incorrect: First part is plausible the EOP does direct going to single RBS pump operation during LOCA mitigation however it does so as a result of valve failures to increase the time you remain in the injection phase and therefore time to address the valve failure. Second part is plausible since it would be correct if the question were regarding RBS system impact on offsite doses.

B. Incorrect: First part is plausible the EOP does direct going to single RBS pump operation during LOCA mitigation however it does so as a result of valve failures to increase the time you remain in the injection phase and therefore time to address the valve failure. Second part is correct in that TSP baskets are credited with keeping Iodine in solution during the recirc phase of a LOCA..

C. Incorrect: First part is correct. Second part is plausible since it would be correct if the question were regarding RBS system impact on offsite doses.

D. CORRECT: TSP baskets are credited with keeping Iodine in solution during the recirc phase of a loca and thereby ensuring offsite dose within limits during DBA.

Technical Reference(s): **SLC 16.6.10 bases CH-CAS**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **ASM-TSS R5**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension and Analysis**

Question 93

Unit 3 initial conditions:

- Reactor power = 100%
- Fuel movement in Unit 3 Spent Fuel Pool

Current conditions:

- 3RIA-6 (SFP) in HIGH alarm
- 3RIA-41 (SFP Gas) in HIGH alarm
- 3SA-8/B9 (PROCESS MONITOR RADIATION HIGH) in alarm

Based on the above conditions, which ONE of the following describes the Abnormal Procedure of the highest priority and the Outside Air Booster Fans that will be started as part of the mitigating actions?

A. AP/9 (Spent Fuel Damage) / Unit 3 Outside Air Booster Fans ONLY

B. AP/9 (Spent Fuel Damage) / Unit 1 and 2 AND Unit 3 Outside Air Booster Fans

C. AP/18 (Abnormal Release of Radioactivity) / Unit 3 Outside Air Booster Fans ONLY

D. AP/18 (Abnormal Release of Radioactivity) / Unit 1 and 2 AND Unit 3 Outside Air Booster Fans

Question 93
T2/G2 -cpw

072G2.4.4, Area Radiation Monitoring
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (4.5/4.7)

K/A MATCH ANALYSIS
Requires recognizing RIA-6 as part of entry conditions for AP/9

SRO-ONLY ANALYSIS
Requires recognizing entry conditions for 2 AP's are met and prioritizing which AP should be directed by the PD consistent with OMP 1-18 section 5.3 (Procedure Implementation)

ANSWER CHOICE ANALYSIS
Answer: B

A. Incorrect: First part is correct. Second part is plausible since the fuel handling accident is in the Unit 3 SFP it would be reasonable to deduce that only Unit 3 Outside Air Booster Fans would be needed.

B. **CORRECT: Entry conditions to both AP/9 and AP/18 are met however AP/18 (in this case) contains no actions to control the plant or mitigate the event therefore AP/9 would be the higher priority AP to be directed by the PD. Choosing AP/9 first is correct path to initiated SFVS and OABF's most expeditiously. AP/9 then directs starting both Unit 3 AND Unit 1&2 OABF's.**

C. Incorrect: First part is plausible since entry conditions are met and AP/18 will need to be initiated. Second part is plausible since the fuel handling accident is in the Unit 3 SFP it would be reasonable to deduce that only Unit 3 Outside Air Booster Fans would be needed.

D. Incorrect: First part is plausible since entry conditions are met and AP/18 will need to be initiated. Second part is correct

---

Technical Reference(s): **AP/18 (Abnormal Release of Radioactivity), AP/9 (Spent Fuel Damage)**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-APG R8, R9**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 94

Which ONE of the following describes a condition that would require making a plant page to implement Emergency Dose Limits AND whether to include SG tube leakage when calculating RCS Pressure Boundary leakage (gpm) in accordance with Tech Spec 3.4.13 (RCS Operational Leakage)?

A. 45 gpm SGTR / no

B. 45 gpm SGTR / yes

C. 45 gpm RCS leak / no

D. 45 gpm RCS leak / yes

Question 94

Conduct of Ops- (selection of procedures)

2.1.14, **Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.** (3.1)

K/A MATCH ANALYSIS

Requires knowledge of conditions that require a plant-wide announcement regarding implementation of Emergency Dose Limits

SRO-ONLY ANALYSIS

Requires knowledge from the bases of TS 3.4.13 regarding what constitutes RCS Pressure Boundary Leakage. That info excludes SG tube leakage from being pressure boundary leakage when applying the TS limits on leakage in TS 3.4.13 and that information is only available in the bases of the TS.

ANSWER CHOICE ANALYSIS

Answer: A

A. **CORRECT: EDL's only apply when the EOP has been initiated. Once in the EOP, either a SGTR OR a LOCA will result in EDL's being in affect. The 45 gpm SGTR would require entry into the EOP since it exceeds the 25 gpm threshold value for EOP entry. The 45 gpm RCS leak will be handled from AP/2 (RCS leakage) therefore the SGTR would require EDL's. The bases of TS 3,4,13 (RCS leakage) clearly excludes SG tube leakage from being "Pressure Boundary" leakage.**

B. Incorrect: First part is correct. Second part is plausible since the SG tubes are actually a barrier that withstands RCS pressure however based on specific knowledge contained in the bases of TS 3.4.13, SGTR leakage is excluded from the requirements of Pressure Boundary Leakage.

C. Incorrect: First part is plausible since it is reasonable to assume (and correct to assume) that sufficient RCS leakage would require implementing EDL's however 45 gpm is not enough. The misconception that EDL's apply to any significant RCS leakage (instead of the actual requirement of a LOCA) would result in this choice. Second part is correct

D. Incorrect: First part is plausible since it is reasonable to assume (and correct to assume) that sufficient RCS leakage would require implementing EDL's however 45 gpm is not enough. The misconception that EDL's apply to any significant RCS leakage (instead of the actual requirement of a LOCA) would result in this choice. Second part is plausible since the SG tubes are actually a barrier that withstands RCS pressure however based on specific knowledge contained in the bases of TS 3.4.13, SGTR leakage is excluded from the requirements of Pressure Boundary Leakage.

Technical Reference(s): **OMP-1-2, OMP 2-1, TS 3.4.13**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **ADM-TSS R5**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Knowledge and Fundamentals**

Question 95

Unit 2 initial conditions:

- Refueling in progress
- FTC level = >21.34 feet stable
- No water additions are being made to the system
- 2A LPI train is operable and in service

Current conditions:

- Refueling SRO desires stopping the 2A LPI Pump to aid in inserting a fuel assembly
- 2A LPI pump has been in continuous operation for the previous 24 hours

Based on the above conditions, which ONE of the following describes whether the 2A LPI pump may be stopped in accordance with OP/2/A/1502/007 (Operations Defueling /Refueling Responsibilities) and the Tech Spec bases for requiring only 1 OPERABLE DHR loop in this condition?

A. 2A LPI Pump may be stopped for up to 1 hour per 8 hour period.
FTC level is within TS limits and provides adequate backup decay heat removal.

B. 2A LPI Pump may be stopped for up to 1 hour per 8 hour period.
Spent Fuel Cooling system provides adequate backup decay heat removal.

C. 2A LPI Pump may NOT be stopped
FTC level is below TS limits and cannot provide adequate backup decay heat removal.

D. 2A LPI Pump may NOT be stopped
Spent Fuel Cooling system can NOT provide adequate backup decay heat removal.

Question 95 - CPW

Conduct of Ops-

2.1.41 **Knowledge of the refueling process.** (3.77) 43.6

K/A MATCH ANALYSIS

Requires knowledge of fuel handling procedures and requirements

SRO-ONLY ANALYSIS

Requires knowledge of procedures and limitations involved in core alterations (43.6), TS and Bases (43.2)

ANSWER: A

A. **CORRECT: TS 3.9.4 (Refueling Ops- DHR and Coolant Circulation –High Water Level) is in effect as water level is ≥ 21.34 ft. This condition requires only 1 DHR loop to be operable and in service since the water can provide adequate backup decay heat removal. TS and Refueling procedures limits & precautions allow SRO to grant permission for the operating loop to be secured for up to 1 hour every 8 hours with adequate level.**

B. Incorrect: First part is correct. 2nd part is incorrect but plausible since Spent Fuel Cooling (SFC) helps to provide decay heat removal but is not the basis for allowing the pump to be secured for up to an hour.

C. Incorrect: First part is incorrect but plausible if TS 3.9.5 criteria are misapplied to this situation. TS3.9.5 (DHR and Coolant Circulation – Low Water Level) requires 2 operable DHR loops with one loop in service (no time is allowed for a pump to be secured). 2nd part is wrong but plausible in that it would be true if level was below the refueling level of 21.34 ft.

D. Incorrect: First part is incorrect but plausible if TS 3.9.5 criteria are misapplied to this situation. TS3.9.5 (DHR and Coolant Circulation – Low Water Level) requires 2 operable DHR loops with one loop in service (no time is allowed for a pump to be secured). 2nd part is incorrect but plausible since Spent Fuel Cooling (SFC) helps to provide decay heat removal but is not the basis for allowing the pump to be secured for up to an hour.

Technical Reference(s): **OP/0/A/1506/001, OP/2/A/1502/007, TS 3.9.4, TS 3.9.5**

Proposed references to be provided to applicants during examination: **None**

Learning Objective: **FH-FHS R21, R32 // ADM-TSS R5, R6**

Question Source: **BANK**

Question History: Last NRC Exam **2009 Question 95**

Question Cognitive Level: **Knowledge and Fundamentals**

1 POINT

Question 96

Unit 1 plant conditions:

- OP/1/A/1102/001 (Controlling Procedure for Unit Startup) Enclosure 4.7 (Unit Startup From 532F/2155 psig To MODE 1) in progress
- ECP = 68% on Group 6
- Group 5 = 76% withdrawn
- Control power will NOT turn on to group 6 rods
- Repairs will take about 4 hours

Which ONE of the following…

(1) describes the direction that will be given to the OATC based on the above plant conditions…
AND
(2) the consequences of power operations while in the Unacceptable Operation region of the rod curves in the COLR in accordance with the bases for TS 3.2.1 (Regulating Rods Position Limits)?

A. Insert control rods until all Safety Rods are fully inserted / Required shutdown margin may not exist

B. Insert control rods until all Safety Rods are fully inserted / Quadrant Power Tilt limits may be exceeded

C. Insert control rods to Group 1 at 50% / Required shutdown margin may not exist

D. Insert control rods to Group 1 at 50% / Quadrant Power Tilt limits may be exceeded

Question 96

Equipment Control -cpw

2.2.2, **Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.**
(4.1)

K/A MATCH ANALYSIS

NO 43 tie. Per NRC, OK to ask about SRO decision resulting in directing manipulation of controls (example inserting rods with missed ECP)
Requires directing RO to manipulate console controls during Reactor Startup.

SRO-ONLY ANALYSIS

Requires knowledge of TS bases behind different portions of rod worth curves as described in TS 3.2.1 (Regulating Rods Position Limits) bases

ANSWER CHOICE ANALYSIS

Answer: C

A. Incorrect: First part is plausible since procedure guidance does direct inserting Safety Rods if the startup is delayed however the procedure directs leaving Gp 1 50% withdrawn for available negative reactivity insertion. Second part is correct.
B. Incorrect: First part is plausible since procedure guidance does direct inserting Safety Rods if the startup is delayed however the procedure directs leaving Gp 1 50% withdrawn for available negative reactivity insertion. Second part is plausible since control rod position does affect tilt readings however homogeneous group rod positions generally effect imbalance and not tilt. It would be reasonable to have a misconception that resulting in confusing the tilt vs imbalance impact of rod positions. Additionally, rod position can impact QPT assuming rod misalignment issues exist therefore associating tilt issues with control rod position issues is plausible
C. **CORRECT: Procedure limits and precautions direct inserting rods to Gp 1 at 50% if startup is delayed once approach to criticality had begun. The bases of TS 3.2.1 (Regulating Rods Position Limits) describes the consequences of being in the unacceptable region as potentially not having the required Shutdown Margin.**
D. First part is correct. Second part is plausible since control rod position does affect tilt readings however homogeneous group rod positions generally effect imbalance and not tilt. It would be reasonable to have a misconception that resulting in confusing the tilt vs imbalance impact of rod positions. Additionally, rod position can impact QPT assuming rod misalignment issues exist therefore associating tilt issues with control rod position issues is plausible.

Technical Reference(s): **OP/1/A/1102/001 Encl 4.7, TS 3.2.1 (Regulating Rods Position Limits)**
Proposed references to be provided to applicants during examination: **NONE**
Learning Objective: **CP-011 R37, 38 ADM-TSS R5**
Question Source: **NEW**
Question History: Last NRC Exam **N/A**
Question Cognitive Level: **Knowledge and Fundamentals**

1 POINT

Question 97

Unit 3 plant conditions:

- LPI aligned to normal DHR mode using 3A LPI pump
- 3B LPI train is aligned in its ECCS mode
- ALL RCP's secured
- CETC's = 196°F stable
- RCS Tcold = 205°F stable
- 3A LPI Cooler Outlet temperature = 195°F stable

Based on the above conditions, which ONE of the following describes the current plant MODE AND the number of OPERABLE DHR loops?

A. MODE 4 / 1

B. MODE 4 / 2

C. MODE 5 / 1

D. MODE 5 / 2

Question 97

Equipment Control -

2.2.35

Ability to determine Technical Specification Mode of Operation. (4.5)

K/A MATCH ANALYSIS

Requires analyzing plant data and determining TS MODE of operation

SRO-ONLY ANALYSIS

Requires knowledge from the TS Bases of TS 3.4.7 regarding alignment requirements of an LPI loop when being credited for an operable DHR loop

ANSWER CHOICE ANALYSIS

Answer: D

A. Incorrect: First part is plausible since Tc is > 200°F. With no RCP's operating, RCS temperature is LPI Cooler Outlet temperature. Based on misconception that Tc should be used for RCS temperature the Rx would be in MODE 4. Second part is plausible since the second LPI loop is aligned in the ECCS mode. It would be plausible to have the misconception that the loop not in operation would be an inoperable DHR loop since it would have to be aligned in the DHR mode to be able to provide Decay Heat Removal capability.

B. Incorrect: First part is plausible since Tc is > 200°F. With no RCP's operating, RCS temperature is LPI Cooler Outlet temperature. Based on misconception that Tc should be used for RCS temperature the Rx would be in MODE 4. Second part is correct

C. Incorrect: First part is correct. Second part is plausible since the second LPI loop is aligned in the ECCS mode. It would be plausible to have the misconception that the loop not in operation would be an inoperable DHR loop since it would have to be aligned in the DHR mode to be able to provide Decay Heat Removal capability.

D. CORRECT: Since no RCP's are operating, LPI cooler outlet temp is RCS temp and once RCS temp goes below 200°F MODE 5 is entered. TS 3.4.7 describes operable DHR loop as one that is capable of being manually realigned to the DHR mode of operation therefore both LPI loops are operable DHR loops.

Technical Reference(s): **TS 3.4.6, TS 3.4.7, SR 3.0.3**

Proposed references to be provided to applicants during examination: **TS 3.4.6 & 3.4.7**

Learning Objective: **ADM-ITS R3**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 98

Unit 3 plant conditions:

- Startup in progress
- Lead blankets discovered on 3B HPI injection nozzles
- Engineering determines that the lead blankets render 3B HPI train inoperable
- Planned Special Exposure will be used to remove the lead blankets

Based on the above conditions, which ONE of the following describes…

(1) the maximum exposure (TEDE) to an individual allowed while removing the lead blankets
AND

(2) the maximum Reactor power (%) at which HPI can perform its credited functions with ONE HPI train incapable of providing flow (neither automatically nor manually) in accordance with Tech Spec bases?

A. 5 rem / 100

B. 5 rem / 75

C. 25 rem / 100

D. 25 rem / 75

Question 98

Radiation Control -

2.3.4 **Knowledge of radiation exposure limits under normal or emergency conditions.** (3.7)

K/A MATCH ANALYSIS
Requires knowledge of Planned Special Exposure limits that apply under "exceptional" circumstances.

SRO-ONLY ANALYSIS
Requires knowledge of bases for TS 3.5.2 (HPI)

ANSWER CHOICE ANALYSIS
Answer: B

A. Incorrect: First part is correct. Second part is plausible since the normal situation for safety systems is to be "single failure proof" which generally means there is one train of the system required to perform the safety function and one additional train. This is true for most all ONS safety systems (ex. LPI, RBS, RBCU) however a single train of our HPI system is not capable of mitigating all break sizes and locations of SBLOCA's when above 75% RTP.

B. CORRECT: Planned Special Exposure maximum exposure is 5 rem TEDE in any calendar year therefore 5 rem would be the maximum allowed. If the plant is operating at > 75% RTP, and one HPI train is inoperable and incapable of being automatically actuated or manually aligned from the Control Room to provide flow post-accident, the HPI System would be incapable of performing its safety function.

C. Incorrect: First part is plausible since 25 rem is the total lifetime limit for Planned Special Exposures. Second part is plausible as described in A.

D. Incorrect: First part is plausible since 25 rem is the total lifetime limit for Planned Special Exposures. Second part is correct.

Technical Reference(s): **TS 3.5.2, RAD-RPP**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **ADM-ITS R3, RAD-RPP R7**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Knowledge and Fundamentals**

1 POINT

Question 99

Plant conditions:

- All 3 Units reactor power = 100%
- 1SA-3/B6 (FIRE ALARM) actuated
- NEO's dispatched to the Turbine Building Basement (Unit 1 and 2 Powdex area)

Current conditions:

- NEO reports: Fire location is TBB/K28. Heavy smoke and rolling flames (5-10ft.) spreading out from the Powdex Alarm Panel

Based on the above conditions, which ONE of the following describes the unit(s) requiring a reactor trip AND who is allowed to serve as a Fire Brigade leader in accordance with the bases of SLC 16.13.1 (Minimum Station Staffing Requirements) ?

REFERENCE PROVIDED

A. ALL units / SRO ONLY

B. Unit 2 (only) / SRO ONLY

C. ALL units / SRO OR an NEO with appropriate qualifications

D. Unit 2 (only) / SRO OR an NEO with appropriate qualifications

Question 99
Emerg Procs /E-Plan -

2.4.27 **Knowledge of "fire in the plant" procedures.** (3.9)

K/A MATCH ANALYSIS
Requires knowledge of and ability to utilize procedures related to fires in the plant

SRO-ONLY ANALYSIS
Requires knowledge from the bases of SLC 16.13.1 regarding who can serve as fire brigade leader.

ANSWER CHOICE ANALYSIS
Answer: D

A. Incorrect: First part is plausible since misreading the Attachment in the ARG could result in determining that all units must enter AP/25 and therefore trip their reactors. Second part is plausible since under normal conditions the SRO is the Fire Brigade leader.

B. Incorrect: First part is correct. Second part is plausible since under normal conditions the SRO is the Fire Brigade leader.

C. Incorrect: First part is plausible since misreading the Attachment in the ARG could result in determining that all units must enter AP/25 and therefore trip their reactors. Second part is correct.

D. CORRECT - The ARG defines this fire as a CHALLENGING ACTIVE FIRE. Utilizing Attachment 1 of the ARG you will determine you are inside an SSF Risk area that affects only Unit 2. The ARG directs the affected control room crew to enter AP/25 and AP/25 directs manually tripping the reactor. The bases of SLC 16.13.1 allows for either an SRO or a qualified NEO be the fire brigade leader.

Technical Reference(s): **1SA-3/B6 (FIRE ALARM) , AP/25, SLC 16.13.1 bases**
Proposed references to be provided to applicants during examination: **1SA-3/B6 (FIRE ALARM)**
Learning Objective: **EAP-SSF R10**
Question Source: **NEW**
Question History: Last NRC Exam N/A
Question Cognitive Level: **Comprehension and Analysis**

1 POINT

Question 100

Unit 1 initial conditions:

- Reactor power = 100%
- 1RIA-59 reading 2 gpm slowly increasing
- AP/31 (Primary to Secondary Leakage) in progress

Current conditions:

- Reactor power = 75% decreasing
- 1RIA-59 reading 28 gpm increasing

Based on the above conditions, which ONE of the following describes how the SRO will proceed AND the actions required if the steam generator approaches overfill conditions (285" XSUR) during the cooldown?

A. Continue shutdown under the direction of AP/31 / Drain the SG to the hotwell

B. Continue shutdown under the direction of AP/31 / Steam the SG to the condenser

C. Go directly to SGTR tab / Drain the SG to the hotwell

D. Go directly to SGTR tab / Steam the SG to the condenser

Question 100

Emerg Procs /E-Plan -

2.4.8 **Knowledge of how abnormal operating procedures are used in conjunction with EOPs.** (4.5)

K/A MATCH ANALYSIS

Question requires knowledge of how AP/31 is utilized in conjunction with EOP based on SGTR size.

SRO-ONLY ANALYSIS

Requires assessing plant conditions (approaching SG overfill) and prescribing a section of a procedure with which to proceed (section of EOP tab that directs steaming to prevent overfill).

ANSWER CHOICE ANALYSIS

Answer: D

A. Incorrect: First part is plausible since it is true until the SGTR reaches 25 gpm. Since it is now 28 gpm, transfer to the EOP is required. Second part is plausible since it would be a method considered to reduce SG level if cooldown were occurring with normal plant procedures via AP/31.

B. Incorrect: First part is plausible since it is true until the SGTR reaches 25 gpm. Since it is now 28 gpm, transfer to the EOP is required. Second part is correct.

C. Incorrect: First part is correct. Second part is plausible since it would be a method considered to reduce SG level if cooldown were occurring with normal plant procedures via AP/31.

D. CORRECT: Transfer directly to SGTR tab of the EOP is directed once SGTR reaches 25 gpm. During the cooldown, if SG approaches overfill, SGTR tab directs steaming the affected SG to prevent overfill even if TS cooldown rates are violated.

Technical Reference(s): **EOP-SGTR, AP/31**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: **EAP-SGTR R26 EAP-APG R1, R8**

Question Source: **NEW**

Question History: Last NRC Exam **N/A**

Question Cognitive Level: **Knowledge and Fundamentals**