KA	KA_desc
SYS003	SYS003 GENERIC Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)
2.1.20	

Given the following conditions on Unit 1:

- The unit was initially at 100% RTP
- #1 Seal Leakoff on 1A NC pump indicates 6.5 GPM
- AP-08 (Malfunction of NC Pump) Case I (NC Pump Seal or Pump Lower Bearing Malfunction has been implemented
- The crew has reached the steps in AP-08 to trip the Reactor and stop the 1A NC pump

In accordance with AP-08, Enclosure 2 (NC Pump Post Trip Actions For #1 Seal Failure) must be performed within 3-5 minutes after stopping the 1A NC pump to prevent __(1)__. The requirement to perform these actions is applicable __(2)__.

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

A. 1. damage to the 1A NC pump #2 & #3 seals
   2. only while AP-08 is in effect

B. 1. damage to the 1A NC pump #2 & #3 seals
   2. even after transition from AP-08 to E-0

C. 1. the VCT from exceeding design temperature limits
   2. only while AP-08 is in effect

D. 1. the VCT from exceeding design temperature limits
   2. even after transition from AP-08 to E-0

General Discussion

In accordance with AP-08, after the Reactor is tripped and the NC pump is stopped the seal return valve for the AFFECTED NC pump (1NV-34A) must be closed within 3-5 minutes. This action is contained in Enclosure 2 (NC Pump Post Trip Actions For #1 Seal Failure). In accordance with the AP-08 Background Document, the seal return line must be isolated to prevent damage to the #2 and #3 seals due to high temperature water flowing past the seals.

Per AP-08, the requirement to close 1NV-34A within 3-5 minutes after stopping the pump is applicable even after transition to the EPs.

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is correct.

Part 2 is plausible because conditional steps in APs are typically no longer applicable when transition is made to the EPs. In this particular case the applicant must recall the caution from AP-08 that states the post pump trip actions in the AP-08 enclosure are applicable even after transition to the EPs to arrive at the correct answer.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible because this is the basis for closing 1NV-94AC and 95B (NC Pumps Seal Return Cont Isolations) in AP-08 when both seal injection and thermal barrier cooling are lost.

Part 2 is plausible because conditional steps in APs are typically no longer applicable when transition is made to the EPs. In this particular case the applicant must recall the caution from AP-08 that states the post pump trip actions in the AP-08 enclosure are applicable even after transition to the EPs to arrive at the correct answer.

Answer D Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible because this is the basis for closing 1NV-94AC and 95B (NC Pumps Seal Return Cont Isolations) in AP-08 when both seal injection and thermal barrier cooling are lost.

Part 2 is correct.

Basis for meeting the KA

The applicant demonstrates the ability to interpret procedure steps by demonstrating a knowledge of basis for performing the Post Pump Trip Actions of Enclosure 2 (specifically closing the NC pump seal return valves within 3-5 minutes). The applicant demonstrates the ability to execute procedure steps by demonstrating the knowledge that the AP procedure steps must be performed even after transition to the EPs.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the "Clarification Guidance for SRO-only Questions (Rev 1 dated 03/11/2010)" under the Screening Criteria for question linked to 10CFR55.43(b)(5) (Assessment and Selection of Procedures):

1) The question can NOT be answered by knowing systems knowledge alone. The basis for closing the seal return isolation valve for the affected pump within 3-5 minutes is not covered by the NCP system lesson plan. Therefore, this is not systems level knowledge.
2) The question can NOT be answered by knowing immediate Operator actions.
3) The question can NOT be answered by knowing AOP or EOP entry conditions.
4) The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.
5) The question requires the applicant to recall procedure content from AP-08 (i.e. that the Post Pump Trip Actions must still be performed even after transition to the EPs). Additionally, the applicant must recall why the procedure steps must be performed from the AP-08 Background Document.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

2010 MNS SRO NRC Examination QUESTION 76

2576

B

Development References	Student References Provided
esson Objective: 1) N/A References: 1) AP-08, Malfunction of NC Pump 2) AP-08 Background Document	

KA	KA_desc
SYS003	SYS003 GENERIC□Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)
2.1.20	

401-9 Comments:	Remarks/Status

Question 76 References:

From AP-08:

MNS AP/1/A/5500/08 UNIT 1	MALFUNCTION OF NC PUMP Case I NC Pump Seal or Pump Lower Bearing Malfunction	PAGE NO. 5 of 24 Rev. 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. **Check if seal cooling available to affected pump:** __ • Seal injection established (Normal or SSF supply) OR __ • KC to thermal barrier established.	**Perform the following:** a. CLOSE one of the following: • 1NV-94AC (NC Pumps Seal Ret Cont Inside Isol) OR __ • 1NV-95D (NC Pumps Seal Ret Cont Outside Isol). __ b. **IF AT ANY TIME** seal cooling is restored, **THEN** observe Note prior to Step 7 and **GO TO** Step 7. __ c. **RETURN TO** Step 2.

NOTE Up to 24 hours of NC pump operation may be required before seals seat and operate normally after seal maintenance or startup.

__ 7. **Check any NC pump number 1 seal leakoff - GREATER THAN OR EQUAL TO 6 GPM.**	**Perform the following:** **NOTE** CP/1/A/6200/001B (Chemical and Volume Control System Charging), Enclosure 4.10 (Maintaining NC Pump Seal Leakoff) gives guidance on actions used to change seal leakoff flow. __ a. **IF** seal leakoff slowly going up, **THEN** contact station management for further guidance. __ b. Continue to monitor NC pump seal leakoff flow. __ c. **IF AT ANY TIME** seal leakoff flow goes up to 6 GPM, **THEN GO TO** Step 8. __ d. **GO TO** Step 9

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

8. **Stop affected NC pump as follows:**

   a. **IF** A or B NC pump is the affected pump, **THEN** CLOSE associated spray valve:

   __ • 1NC-27C (A NC Loop PZR Spray Control)

   __ • 1NC-29C (B NC Loop PZR Spray Control).

CAUTION **Enclosure 2 (NC Pump Post Trip Actions For #1 Seal Failure) contains actions that must be performed between 3 and 5 minutes after stopping NC pump. This enclosure must be performed even after transition to EPs.**

__ b. Have any available RO perform Enclosure 2 (NC Pump Post Trip Actions For #1 Seal Failure) as crew performs the following steps.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION **Failure of Number 2 and 3 seals may occur unless the affected NC pump seal return valve is closed between 3 minutes and 5 minutes after stopping pump. This enclosure must be completed even after transition to EPs.**

__ 1. **Record time of NC pump shutdown:**

_____

2. **Check if seal cooling available to affected pump:**

__ • Seal injection established (Normal or SSF Supply)

OR

__ • KC to thermal barrier established.

Perform the following:

a. CLOSE the following:

__ • 1NV-94AC (NC Pumps Seal Ret Cont Inside Isol)

__ • 1NV-95B (NC Pumps Seal Ret Cont Outside Isol).

__ b. Exit this enclosure.

__ 3. **Check any NC pump number 1 seal leakoff flow - GREATER THAN OR EQUAL TO 6 GPM.**

__ **GO TO Step 5.**

__ 4. **Maintain seal injection flow greater than 9 GPM to affected pump(s).**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **WHEN affected NC pump has been off 3 minutes, THEN immediately perform the following:** a. CLOSE affected NC pump seal return valve: __ • 1NV-34A (A NC Pump Seal Return Isol) __ • 1NV-50B (B NC Pump Seal Return Isol) __ • 1NV-66A (C NC Pump Seal Return Isol) __ • 1NV-82B (D NC Pump Seal Return Isol). b. OPEN all of the following valves: __ • OPEN 1KC-394A (A NC Pump Therm Bar Otlt). __ • OPEN 1KC-345A (C NC Pump Therm Bar Otlt). __ • OPEN 1KC-364B (B NC Pump Therm Bar Otlt). __ • OPEN 1KC-413B (D NC Pump Therm Bar Otlt).	

From AP-08 Background Document:

CASE I STEP 6:

PURPOSE:

Prevent hot NC Pump seal return flow from going to the VCT and prevent transition to the steps that may close individual pump seal return valves if no seal cooling exists.

DISCUSSION:

With no seal injection coincident with no thermal barrier cooling, this step closes NV-94 and 95 (NC Pump seal return containment isolations). This will force the hot #1 seal leak-off flow to the PRT and prevent the VCT from exceeding design temperature limits (150 F). The RNO of this step will also prevent transition to the next several steps dealing with specific seal failures, and back to monitoring for NCP trip criteria. This has the benefit of skipping the steps that would close the individual seal return isolation valves. The individual seal return isolation valves need to remain open for loss of all seal cooling events. The following is an excerpt from DW-94-011:

Isolation of the #1 seal leakoff line during a loss of all seal cooling event would force the #2 NCP seal into the high pressure mode of operation at high temperature. This is beyond the design basis of the #2 seal, and the response of the #2 seal to high pressure operation without cooling is unknown. The analysis performed for the extended loss of ac power in WCAP-10541, Rev. 2, identifies that "the high temperature two-phase flow through the seal system and #1 seal leakoff line increases the pressure in the #1 leakoff cavity. The increased leakoff cavity pressure tends to decrease the separation of the #1 seal faces which tend to reduce the leakage. The combination of competing effects results in higher leakage rates, but leakage rates that are self-limiting. Higher flow increases the system back pressure which reduces the separation of the faces, reducing the flow." Therefore, keeping the #1 seal leakoff line open will provide the benefit of minimizing the leakage, while closing the leakoff line could result in catastrophic failure of the #2 seal and actually increase leakage.

In summary, closing NV-94 & 95 keeps individual flowpath open for #1 seal leakoff (through relief to PRT), but isolates it from the VCT.

Note: this AP does not address restoring a loss of seal injection or thermal barrier cooling. Other APs address those problems.

REFERENCES:
DW-94-011 page 6 of 7
WCAP-10541, Rev. 2

CASE I STEP 8:

PURPOSE:

Provide the direction for stopping the affected NC Pump.

DISCUSSION:

Closing a spray valve for "A" or "B" pump ensures Pzr pressure control is maintained. The operator is cued to close the spray valve in anticipation of losing the motive force for flow through the valve. The operator is expected to take whatever other compensatory action is required to stabilize Pressurizer pressure (operating heaters or other spray flow).

Between three and five minutes after the pump has been tripped, the affected pumps seal return isolation is closed. Waiting three minutes ensures the pump has stopped rotating. The #2 seal has a softer seat than the #1 seal, and if rotating while exposed to the potential debris from the failed #1 seal, it could experience premature failure. Closing it in less than 5 minutes minimizes the time the #2 seal is exposed to high temperature fluid conditions.

Closing the seal return valve within 3 to 5 minutes during a #1 seal failure event does not meet the criteria of "high PRA values" as determined by the Severe Accident Analysis Group. For McGuire, this corresponds to a Risk Achievement Worth (RAW) greater than or equal to 1.04. PIP M-03-1992 documents the events that meet these criteria. As such this action is not a McGuire 'time critical action', but is a management expectation and prudent action to prevent damage to the #2 and #3 seals. PIP M-07-0310 ACA#4 documents the removal of this action from McGuire's 'time critical action' list.

After the affected seal return is closed, the thermal barrier outlet valve is opened, if necessary. This is after the previous step to ensure it's after any perturbations that would close the valve.

An available RO is designated to perform Enclosure 2 (NC Pump Post Trip Actions for #1 Seal Failure). This could be an extra RO if available. If one is not available, this could be the BOP. It has to be someone. The use of an enclosure facilitates the designated person completing the enclosure actions while minimizing the interaction with the crew. The enclosure can be handed off to the designated person while the crew can focus on just E-0. With an enclosure, additional communications between the RO and the SRO aren't required to complete these actions.

If in Mode 1 or 2, the guidance for stopping a NC Pump includes: tripping the reactor, waiting for reactor power to decay below 5%, tripping the affected NC pump, and transition to E-0. In Mode 3, 4, or 5, this is not necessary. This is why the two steps for stopping an NCP are written differently.

Guidance is given to wait until reactor power is less than 5% before stopping the NC pump. This will ensure the NC pump will provide adequate flow/core cooling until reactor power is sufficiently low enough to preclude a challenge to fuel integrity.

KA	KA_desc
SYS005 A2.02	Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)□Pressure transient protection during cold shutdown .....

Given the following conditions on Unit 1:

- The unit is in Mode 5
- NC system temperature is currently 112°F
- 1A ND Train is in service
- A special test procedure is to be run which requires BOTH NI pumps to be run in parallel and aligned to inject into the NC system.

Which ONE (1) of the following describes the requirements per Tech Spec 3.4.12 (LTOP) Bases?

A. Secure two PORVs open with associated block valves open and power removed.
This action protects against brittle fracture due to pressurized thermal shock of the reactor vessel.

B. Secure two PORVs open with associated block valves open and power removed.
This action protects against brittle fracture due to cold overpressure of the reactor vessel.

C. Establish an RCS vent of ≥ 2.75 square inches and verify at least ONE Operable PZR PORV.
This action protects against brittle fracture due to pressurized thermal shock of the reactor vessel.

D. Establish an RCS vent of ≥ 2.75 square inches and verify at least ONE Operable PZR PORV.
This action protects against brittle fracture due to cold overpressure of the reactor vessel.

General Discussion

Based on the conditions given, the applicant is placed in a condition that if the test is to be run, certain conditions must be met to satisfy the LTOP vent path requirements.

One method of meeting the vent path requirements is to establish an adequate vent path prior to starting the test. For this case, securing open two Pressurizer PORVs or establishing a vent path of greater than or equal to 4.5" will meet those requirements.

Another method of meeting the vent path requirements is establish a RCS vent path of > 2.75 " AND two Operable PORV's.

The second part of the question deals with the basis of LTOP which is the protection of the reactor vessel from brittle fracture at lower temperatures.

Answer A Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is correct and therefore plausible.

Part 2 is plausible because pressurized thermal shock is a low temperature brittle fracture event but is predicated by a rapid overcooling event which sets up a temperature gradient across the reactor vessel. One of the actions which would allow for this test is the verification of an operable RHR suction relief and that NCS temperature is greater than 74° and Cool Down rate <20 Deg/hr. The applicant could misinterpret this as preventing a PTS type event.

Answer B Discussion

CORRECT: See explanation above

Answer C Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because it is partially correct in that the actions stated would meet LTOP requirements but an additional operable PZR PORV would required.

Part 2 is plausible because pressurized thermal shock is a low temperature brittle fracture event but is predicated by a rapid overcooling event which sets up a temperature gradient across the reactor vessel. One of the actions which would allow for this test is the verification of an operable RHR suction relief and that NCS temperature is greater than 74° and Cool Down rate <20 Deg/hr. The applicant could misinterpret this as preventing a PTS type event.

Answer D Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because it is partially correct in that the actions stated would meet LTOP requirements but an additional operable PZR PORV would required.

Part 2 is correct and therefore plausible.

Basis for meeting the KA

The KA is matched because an operation is about to occur (operation of two NI pumps at the same time) that would result in a pressure transient in the RCS during cold shutdown. The applicant is asked to predict the possible impacts (Brittle fracture) and determine the requirements to mitigate the possible consequences of the proposed test. How this test will impact the LTOP system vent path requirements (Pressure transient protection during cold shutdown). The applicant must determine the actions required by Tech Specs (use procedures to control) that will allow both NI pumps to be run simultaneously.

Basis for Hi Cog

This is a higher cognitive level question because it requires multiple mental steps. The applicant must first analyze the given information to determine that the vent path requirements have changed from 2.75" to 4.5". The applicant must then recall from memory all combinations of equiment that would meet the 4.5" vent path requirement.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the "Clarification Guidance for SRO-only Questions (Rev 1 dated 03/11/2010)" under the Screening Criteria for question linked to 10CFR55.43(b)(2) (Tech Specs):

1) It can NOT be answered solely by knowing < 1 hour Tech Specs.
2) It can NOT be answered solely by knowing the LCO/TRM information listed "above-the-line".
3) It can NOT be answered by knowing the Tech Spec Safety Limits or their bases
4) It requires the applicant to have detailed knowledge of Tech Spec 3.4.12 vent path requirements and information from the TS 3.4.12 Basis Document to determine the correct answer.

2010 MNS SRO NRC Examination QUESTION 77

2577

B

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References	Student References Provided
Learning Objectives: 1) PS-NC #24 References: Tech Sped 3.4.12 Basis	

KA	KA_desc
SYS005 A2.02	Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)□Pressure transient protection during cold shutdown .....

401-9 Comments:	Remarks/Status

Question 77 References:

From OP-MC-PS-NC Objectives

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
20	Describe how NCS temperature, pressure, flow and Pzr level are measured and indicated.		X	X	X	
21	Describe the operation and indication readout of the following NCS level instrumentation: • Ultrasonic level detection • WR level • NR level • Sightglass		X	X	X	X
22	State the nominal values for NC System pressure, Th, Tc, Tave, Pzr temperature for Hot Zero Power and Hot Full Power.	X	X	X	X	
23	Given a Limit and/or Precaution associated with the NC System, discuss its basis and when it applies.		X	X	X	X
24	Concerning the Technical Specifications related to the NC System: • Given the LCO title, state the LCO (including any COLR values) and applicability. • For any LCO's that have action required within one hour, state the action. • Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is(are) not met and any actions(s) required within one hour. • Given a set of parameter values or system conditions and the appropriate Tech Spec, determine required action(s). • Discuss the bases for a given Tech. Spec. LCO or Safety Limit. * SRO ONLY			X X X X	X X X X X	X X X X *

From TS 3.4.12 Basis:

BASES

APPLICABILITY (continued)

the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 300°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable LTOP system. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1, A.2.1, A.2.2.1, A.2.2.2, A.3, A.4, A.5.1, and A.5.2

With two centrifugal charging pumps, safety injection pumps, or a combination of each, capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition. Two pumps may be capable of injecting into the RCS provided the RHR suction relief valve is OPERABLE with:

Correct answer, one option to allow the test would be to secure open 2 PORV's

1. RCS cold leg temperature > 174°F (Unit 1), or
2. RCS cold leg temperature > 89°F (Unit 2), or
3. RCS cold leg temperature > 74°F and cooldown rate < 20°F/hr (Unit 1), or
4. RCS cold leg temperature > 74°F and cooldown rate < 60°F/hr (Unit 2), or
5. two PORV's secured open with associated block valves open and power removed, or
6. a RCS vent of ≥ 4.5 square inches, or

ACTIONS (continued)

7. a RCS vent of ≥ 2.75 square inches and two OPERABLE PORVs (the RCS vent shall not be one of the two OPERABLE PORVs).

This provides the plausibility for distracters A and B.

For cases where no reactor coolant pumps are in operation, RCS cold leg temperature limits are to be met by monitoring of BOTH the WR Cold Leg temperatures and Residual Heat Removal Heat Exchanger discharge temperature. With both PORVS and block valves secured open, or with an RCS vent of 4.5 square inches, there are no credible single failures to limit the flow relief capacity. For the RHR relief valve to be OPERABLE, the RHR suction isolation valves must be open and the relief valve setpoint at 450 psig consistent with the safety analysis. The RHR suction relief valves are spring loaded, bellows type water relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

Required Action A.1 is modified by a Note that permits two centrifugal charging pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.

B.1, C.1, and C.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 300°F, an accumulator pressure of 639 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

D.1

In MODE 4 when any RCS cold leg temperature is ≤ 300°F, with one PORV inoperable, the PORV must be restored to OPERABLE status

KA	KA_desc
SYS061 A2.06	Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)□Back leakage of MFW .....

Given the following conditions on Unit 1:

- The unit is at 12% RTP preparing to roll the main turbine
- M1A1276 (U1 CA Temp at Chk Vlv 1CA-37) alarms on the OAC
- 1CA-37 (#1 TD CA to S/G D)

Based on the above conditions:

1. What method would FIRST be used to reduce the temperature at the check valve?

2. How would this action affect the operability of the TD CA Pump?

A. 1. Close 1CA-36 AB (U1 TD CA Pump Disch to 1D S/G Control) and monitor temperature for 15 min.
   2. The U-1 TD CA Pump remains OPERABLE.

B. 1. Close 1CA-36 AB (U1 TD CA Pump Disch to 1D S/G Control) and monitor temperature for 15 min.
   2. The U-1 TD CA Pump shall be declared INOPERABLE.

C. 1. Close 1CA-38B (U1 TD CA Pump Disch to 1D S/G Isol) and start the TD CA pump aligned for recirculation to the UST.
   2. The U-1 TD CA Pump remains OPERABLE.

D. 1. Close 1CA-38B (U1 TD CA Pump Disch to 1D S/G Isol) and start the TD CA pump aligned for recirculation to the UST.
   2. The U-1 TD CA Pump shall be declared INOPERABLE.

General Discussion

The consequence of the situation described would be overheating of the TD CA pump discharge piping which could lead to voiding and ultimately steam binding associated with this pump. The correct response to the alarm associated with OAC point M1A1276 is to reduce CA system piping temperature per OP/1/A/6250/002. Enclosure 4.4 of this procedure directs the operators to first close the control valve on the affected line, which in this case would be 1CA-36AB or the D S/G. If this is unsuccessful, then the pump is run in recirc to cool the discharge line but all of the remaining motor operated control valves would have to be closed first and this would only be done if the closure of the single control valve was not successful. The stem of the question asked for the FIRST action.
The operability of the TD CA pump is affected both by the closure of the Air Operated flow control valves (1CA-36 AB). Above 10% RTP, closing this valve renders the pump inoperable.

Answer A Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is correct and therefore plausible.

Part 2 is plausible because it the unit was below 10% RTP the action of closing the control valve would not affect the operability of the associated AFW pump. This answer is plausible because it is possible to close this valve with the unit at power without affecting operability just not at the given power level.

Answer B Discussion

CORRECT: See explanation above.

Answer C Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 of the answer is plausible because this action is correct but the stem of the question asked for what method would be used First. The method described would only be employed if the closure of the control isolation was not successful but since it is a possible strategy, it is plausible.

Part 2 is plausible because it the unit was below 10% RTP the action of closing the control valve would not affect the operability of the associated AFW pump. This answer is plausible because it is possible to close this valve with the unit at power without affecting operability just not at the given power level.

Answer D Discussion

INCORRECT: See explanation above,

PLAUSIBLE: Part 1 of the answer is plausible because this action is correct but the stem of the question asked for what method would be used First. The method described would only be employed if the closure of the control isolation was not successful but since it is a possible strategy, it is plausible.

Part 2 is correct and therefore plausible.

Basis for meeting the KA

Part 2 of this question matches the 'a' part of the KA regarding "predict the impact of the following malfunctions on the AFW". The impact is whether the TD CA pump will remain operable.

Part 1 of this question matches the part 'b' of the KA regarding "using procedures to correct, control, or mitigate the consequences". The procedure in this case is OP/1/A/6250/002, Auxiliary Feedwater System, Enclosure 4.5, Reducing Turbine Driven CA Pump Piping Temperature.

Basis for Hi Cog

This question is Hi Cog because the applicant must evaluate a given set of conditions and through a multipart mental process, determine the required actions based on these conditions. The applicant must futher evaluate the impact of the actions to address the high temperature on the operability of the associated AFW pump.

Basis for SRO only

Part 1 of the question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge. Either of these methods can be used procedurally to cool the TD CA pump piping. Check valve leakage is discussed in the systems lesson plan and the methods to cooldown the TD CA pump are mentioned in general terms (i.e. "close the discharge valve or start the pump"). However, the applicant must have detailed knowledge of the OP to discriminate which method is used FIRST. Since this is an infrequently performed evolution, the actions in the procedure are directed by the CR SRO and not handed off to an RO.
2) The question can NOT be answered by knowing immediate operator actions. None of the actions described are immediate actions.
3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs. These are detailed procedure steps from an infrequently performed OP.

4) The question can NOT be answered solely by knowing the purpose, overal sequence of events, or overall mitigative strategy of the procedure. This is detailed knowledge of procedure step sequence not sequence of events within the procedure.
The question requires detailed knowledge of procedure content. Therefore, it is SRO knowledge.

Part 2 of the question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs
2) This question can NOT be answered by knowing information listed "above-the-line".
3) This question can NOT be answered by knowing the TS Safety Limits.
4) This question required the applicant to analyze the given conditions and make the determination that the TD CA pump is inoperable. The applicant must then recall from memory that the unit can not enter MODE 1 with the TD CA pump INOPERABLE.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References	Student References Provided
TS 3.7.5 OP/1/A/6250/002 Auxiliary Feedwater System	

KA	KA_desc
SYS061	Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)□Back leakage of MFW .....
A2.06	

401-9 Comments:	Remarks/Status

Question 78 References:

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

----------NOTE----------
Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

----------NOTE----------
LCO 3.0.4.b is not applicable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	----------NOTE---------- Not applicable to automatic valves when THERMAL POWER is < 10% RTP. ---------- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.3	----------NOTE---------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. ---------- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months

From OP/1/A/6250/002, Auxiliary Feedwater System:

1. Limits and Precautions

None

2. Initial Conditions

_____ 2.1 BW System isolated from S/Gs per OP/1/A/6100/SO-5A (B, C, D) (Draining S/G 1A, 1B, 1C, 1D).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

_____ SRO 3.2 Declare #1 TD CA Pump inoperable.

3.3 Close control valve on affected lines:

_____ CV _____ • 1CA-64AB (U1 TD CA Pump Disch to 1A S/G Control)

_____ CV _____ • 1CA-52AB (U1 TD CA Pump Disch to 1B S/G Control)

_____ CV _____ • 1CA-48AB (U1 TD CA Pump Disch to 1C S/G Control)

_____ CV _____ • 1CA-36AB (U1 TD CA Pump Disch to 1D S/G Control)

☐ 3.4 Monitor temperature for 15 - 30 minutes.

_____ 3.5 IF temperatures remain high after 15 - 30 minutes, close isolation valve on affected lines:

_____ CV _____ • 1CA-66AC (U1 TD CA Pump Disch to 1A S/G Isol)

_____ CV _____ • 1CA-54AC (U1 TD CA Pump Disch to 1B S/G Isol)

_____ CV _____ • 1CA-50B (U1 TD CA Pump Disch to 1C S/G Isol)

_____ CV _____ • 1CA-38B (U1 TD CA Pump Disch to 1D S/G Isol)

Unit 1

NOTE:	When opening valves 1CA-36, 48, 52, and 64 from the local panel, the controller needs to be opened 4 - 5 more turns once 100% is reached to minimize the amount that the valves drift close and back open upon returning controller back to control room (A-Remote).

3.6 **AFTER** temperatures have returned to normal, ensure open:

_____ CV _____ • 1CA-64AB (U1 TD CA Pump Disch to 1A S/G Control)

_____ CV _____ • 1CA-52AB (U1 TD CA Pump Disch to 1B S/G Control)

_____ CV _____ • 1CA-48AB (U1 TD CA Pump Disch to 1C S/G Control)

_____ CV _____ • 1CA-36AB (U1 TD CA Pump Disch to 1D S/G Control)

_____ CV _____ • 1CA-66AC (U1 TD CA Pump Disch to 1A S/G Isol)

_____ CV _____ • 1CA-54AC (U1 TD CA Pump Disch to 1B S/G Isol)

_____ CV _____ • 1CA-50B (U1 TD CA Pump Disch to 1C S/G Isol)

_____ CV _____ • 1CA-38B (U1 TD CA Pump Disch to 1D S/G Isol)

3.7 Check the following stable:

- ☐ M1A1439 (U1 CA Temp at Chk Vlv 1CA-65)
- ☐ M1A1421 (U1 CA Temp at Chk Vlv 1CA-53)
- ☐ M1A1294 (U1 CA Temp at Chk Vlv 1CA-49)
- ☐ M1A1276 (U1 CA Temp at Chk Vlv 1CA-37)

_____ 3.8 **IF** increasing temperatures indicates check valve leak by, perform the following:

_____ 3.8.1 Notify System Engineer.

_____ _____/_____
Person Notified Date Time

_____ 3.8.2 Evaluate operating CA Pumps to cool CA System piping.

3.9 Ensure "TURB" released on the following:

_____ • CA Modulating Valves Reset Train A
_____ • CA Modulating Valves Reset Train B

_____ 3.10 Evaluate operability of CA System.
SRO

End of Enclosure

KA	KA_desc
SYS076 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: (CFR: 41.5 / 43.5 / 45/3 / 45/13)□Loss of SWS .....

Given the following plant conditions:

- A loss of the Low Level Intake has occurred
- AP-20 (Loss of RN) Case II (Loss of Low Level or RC Supply Crossover) has been implemented
- RN is now aligned to the SNSWP
- Train 'B' of RN is now in service on both units

1) In accordance with AP-20, Train 'B' RN pump flow is limited to less than _____.

2) The RN pump flow is limited to prevent _____.

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

A. 1. 11,500 GPM
   2. RN pump impeller wear

B. 1. 13,000 GPM
   2. RN pump impeller wear

C. 1. 11,500 GPM
   2. vibration damage to the KC HX tubes

D. 1. 13,000 GPM
   2. vibration damage to the KC HX tubes

General Discussion

In accordance with AP-20, Loss of RN Case II, Loss of Low Level or RC Crossover, the limit for Train 'A' RN pump flow while aligned to the SNSWP is limited to less than 11,500 GPM and Train 'B' RN pump flow is limited to less than 13, 000 GPM.

In accordance with the AP-20 Background Document, RN pump flow is limited to ensure NPSH is sufficient to prevent long-term RN pump impeller wear.

Answer A Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because this is the flow restriction for Train 'A' RN pumps while aligned to the SNSWP.

Part 2 is correct and is therefore plausible.

Answer B Discussion

CORRECT. As explained above.

Answer C Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because this is the flow restriction for Train 'A' RN pumps while aligned to the SNSWP.

Part 2 is plausible because there are RN flow restrictions for the KC Hx to prevent vibration induced damage to the Hx tubes. However, this is not the basis for the flow restrictions on the RN pumps.

Answer D Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is correct and therefore plausible.

art 2 is plausible because there are RN flow restrictions for the KC Hx to prevent vibration induced damage to the Hx tubes. However, this is not the basis for the flow restrictions on the RN pumps.

Basis for meeting the KA

The K/A is matched because the unit has experienced a loss of all SWS as described in the stem of the question. The system has been placed in an alternate alignment as a result of this event (i.e. now taking suction from the SNSWP instead of the LLI).

Part B of the KA (use procedures to correct control or mitigate) is met by the applicant having knowledge of the AP-20 restrictions on pump flow when aligned to the SNSWP. Part A of the KA (predict the impact) is met by the applicant having knowledge of the potential impact of exceeding the flow limits specified in AP-20.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the "Clarification Guidance for SRO-only Questions (Rev 1 dated 03/11/2010)" under the Screening Criteria for question linked to 10CFR55.43(b)(5) (Assessment and Selection of Procedures):

1) The question can NOT be answered by knowing systems knowledge. The limits for RN pump flow while aligned to the SNSWP are NOT specified in the RN System lesson plan and are NOT specified in the Operating Procedure (OP/1 or 2/A/6400/006, Nuclear Service Water System). The only place these limits are specified is in AP-20. Additionally, the basis for the RN pump flow limits is NOT specified in the RN System Lesson Plan or the Operating Procedure. The basis for the flow restriction is only contained in the AP-20 Background Document. Therefore, neither part of this question is systems level knowledge.
2) The question can NOT be answered by knowing immediate Operator actions.
3) The question can NOT be answered by knowing entry conditions for the AP.
4) The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of AP-20.
5) The question requires the applicant to have knowledge of specific steps within AP-20 which are actions that are directed by the SRO during performance of the AP.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

2010 MNS SRO NRC Examination QUESTION 79

2579

B

Development References	Student References Provided
arning Objective: 1) AP20002 References: 1) AP-20 2) AP-20 Background Document	

KA	KA_desc
SYS076 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: (CFR: 41.5 / 43.5 / 45/3 / 45/13)□Loss of SWS .....

401-9 Comments:	Remarks/Status

Question 79 References:

From AP-20:

MNS AP/1/A/5500/20 UNIT 1	LOSS OF RN Case II Loss of Low Level or RC Supply Crossover	PAGE NO. 47 of 111 Rev. 29

ACTION/EXPECTED RESPONSE		RESPONSE NOT OBTAINED

21. **Maintain RN flow within operating limits as follows:**

- On Unit 1:

__ • IF 1A RN pump is running, THEN THROTTLE 1RN-89A (RN to A KC Hx Control) to maintain 1A RN pump flow less than 11,500 GPM.

__ • IF 1B RN pump is running, THEN THROTTLE 1RN-190B (RN To B KC Hx Control) to maintain 1B RN pump flow less than 13,000 GPM.

- On Unit 2:

__ • IF 2A RN pump is running, THEN THROTTLE 2RN-89A (RN to A KC Hx Control) to maintain 2A RN pump flow less than 11,500 GPM.

• IF 2B RN pump is running, THEN THROTTLE 2RN-190B (RN To B KC Hx Control) to maintain 2B RN pump flow less than 13,000 GPM.

NOTE LLI is the only available suction for RV pumps.

__ 22. **IF RV pumps have lost suction, THEN ensure all RV pumps are in manual and off.**

__ 23. **Ensure Control Room Area Chiller in service PER Enclosure 4 (VC/YC Operation).**

From AP-20 Background Document:

CASE II STEP 21:

PURPOSE:

Maintain the RN flow rate restrictions necessary to ensure NPSH requirements are met on the Pond suction source for the RN Pump (PIP 94-1429). Note short term operations a little above these NPSH flow rates is not an immediate concern, but rather a long term impeller wear issue.

DISCUSSION:

Refer to the below text from System Engineer John Pring from e-mail dated 3/22/00:

"The numbers provided to you in the past by Jim are still accurate, but some clarification appears to be required. These flow limitations are based on **"engineering judgement"** following evaluation of system testing (either flow balance data or pump perfomance testing). The flow limitations are as follows:

Maximum "Suggested" RN Pump flow rate (any pump / any train) from Low Level Intake -- 16,000 gpm

Maximum "Suggested" RN Pump flow rate (any A train pump) from SNSWP -- 11,500 gpm

Maximum "Suggested" RN Pump flow rate (any B train pump) from SNSWP -- 13,000 gpm

The term **"Suggested"** has real meaning in this context. If under emergency or abnormal conditions it becomes necessary to achieve higher flow rates, then the operators should attempt to obtain the required flows. **"Suggested"** means that minimal operating problems will be encountered if the flow thresholds are respected.

Please note that these flow rate limitations are "pump" related. Total Train flow rates will be significantly higher. You can use this note as documentation for the listed pump flow rate limitations.

John"

REFERENCES:
PIP 94-1429 problem evaluation, remarks section & John Pring e-mail 3/22/00

KA	KA_desc
SYS063	SYS063 GENERIC Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)
2.1.23	

With both Units at 100% RTP the following occurs:

- Loss of Offsite Power occurred on Unit 1
- Both DGs started and loaded as designed
- 1A D/G subsequently trips on overspeed
- At Step 17 of ES 0.1 (Reactor Trip Response), the decision is made to implement AP-07 (Loss of Electrical Power)

Which ONE (1) of the following describes the Time Critical actions directed by the CRS to mitigate this event per AP-07?

A. Complete Enc. 7 (DC Bus Alignment) to realign Battery Charger EVCA to Unit 2 within one hour.

B. Complete Enc. 7 (DC Bus Alignment) to realign Battery EVCA to Battery Charger EVCS within one hour.

C. Complete Generic Enc. 13 (VC and VA System Operation) to restart the Train A VC/YC Chiller within 37.5 minutes.

D. Complete Generic Enc. 13 (VC and VA System Operation) to swap Train A VC/YC Chiller power and water to Unit 2 and restart chiller within 1 hour and 15 minutes.

General Discussion

In the scenario described, Battery Charger EVCA would be off due to the tripped D/G 1A. Step 20 directs the implementation of AP-07 Enc. 7 which provides direction for local actions to align EVCA Battery Charger to U-2 and restart the charger. This is required to be complete within one hour to prevent loss of the battery which is designed to provide power for one hour.

Answer A Discussion

CORRECT: See explanation above

Answer B Discussion

INCORRECT: See explanation aboveBattery charger EVCS would only be utilized if it was aligned to the battery prior to the event. This would only be true if Battery Charger was out of service which is not indicted in the initial conditions.

PLAUSIBLE: Because it is the correct Enclosure, time frame and action if EVCS was in service. EVCS charger can be powered from either unit and be aligned to any vital battery so without familiarity with this enclosure this would seem a logical course of action.

Answer C Discussion

INCORRECT: See explanation above. The Train A VC/YC Chiller is normally powered from U-1 "A" Train Vital bus which is deenergized due to the failed 1A D/G.

PLAUSIBLE: Because it is the correct enclosure, correct action and Time requirement if the A VC/YC was available. This chiller can be powered from U-2 but is normally aligned to U-1. The 'time critical' for manually selecting the other chiller if a VC chiller fails is 37.5 minutes.

Answer D Discussion

INCORRECT: See explanation above. See explanation above.Train A VC/YC power and water only need to be swapped if a station blackout has occurred and that both D/G's on one unit fail.

PLAUSIBLE: Because Step 17 of AP-17 directs implementation of this Enclosure within 30 min and should the event be consistent with the need to perform this action, the enclosure and time requirements are correct. If a station blackout occurs and a VC chiller cannot be started due to a loss of power to the chiller, its power supply must be swapped to the opposite unit and the chiller started within 1 hour and 15 minutes from the time of the blackout.

asis for meeting the KA

K/A is matched the question is addressing actions required to recover from a reactor trip due to a LOOP associated with Unit 1 as directed by ES 0.1 and AP-7. In order to successfully answer the question the candidate is required to have a detailed integrated plant procedure knowledge associated with the DC electrical distribution system as well as knowledge of local Time Critical operator actions directed by these procedures along with the associated operational implications of not performing those actions within the given time constraints.

Basis for Hi Cog

The analysis cog level is justified because the candidate must evaluate a given plant scenario, determine equipment availability and using procedural knowledge, determine the required course of action.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the "Clarification Guidance for SRO-only Questions (Rev 1 dated 03/11/2010)" under the Screening Criteria for question linked to 10CFR55.43(b)(5) (Assessment and Selection of Procedures):

1) The question can NOT be answered by knowing systems knowledge alone.
2) The question can NOT be answered by knowing immediate Operator actions.
3) The question can NOT be answered by knowing AOP or EOP entry conditions.
4) The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.
5) The question requires the applicant to assess plant conditions and recall specific procedure content from AP-07. It requires the applicant to have an understanding of the specific procedural requirements associated with two different enclosures and the associated basis for those actions. Therefore, this is an SRO level question.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	MNS Bank NRC Q100

Development References

P/07, Loss of Electrical Power (Rev 28) page 9
-07, Loss of Electrical Power Bkgd (Rev 7)
Pgs 14 & 15

OP-MC-AP-07 Obj: 2

Student References Provided

2010 MNS SRO NRC Examination QUESTION 80

2580

A

KA	KA_desc
SYS063 2.1.23	SYS063 GENERIC Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)

401-9 Comments:	Remarks/Status

Question 80 References:

OP-MC-AP-07 Obj: 3

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Concerning AP/0/5500/7 (Loss of Electrical Power): • State the purpose of the AP • Recognize the symptoms that would require implementation of the AP. AP7001			X	X	X
2	Given scenarios describing accident events and plant conditions, evaluate the basis for any caution, note, or step. AP7002			X	X	X
3	State from memory the Immediate Action(s) and the Response Not Obtained (RNO). SC002			X	X	X

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
CAUTION **If operating train of VC/YC has failed, it may be time critical to swap operating VC/YC trains.**	
__ 17. **Initiate EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 13 (VC And VA System Operation) within 30 minutes of B/O.**	
__ 18. **Check S/G Pressures - STABLE OR GOING UP.**	**IF AT ANY TIME S/G pressure goes down in an uncontrolled manner AND reactor tripped, THEN CLOSE the following valves:** __ • All MSIVs __ • All MSIV bypass valves.

From AP-07 Bkgd Doc (Loss of Electrical Power) Pg 13

UNIT 1 CASE I STEP 17:
UNIT 2 CASE I STEP 15:

PURPOSE:

Ensure control room equipment temperature habitability is maintained.

DISCUSSION:

Excessive control room ambient temperatures could lead to redundant vital control room equipment failures. If the selected train chiller fails (or it' power supply), the opposite chiller doesn't auto start. It's calculated in this type scenario if the opposite train chiller is started (must be done manually) within 37.5 minutes from the onset of the blackout, then equipment habitability will be maintained. If the enclosure is initiated within 30 minutes, then the few minutes it takes to diagnose the failure of the selected train, swap trains, and manually start the opposite train chiller can be accomplished in less than 7.5 minutes.

Other ventilation concerns (containment cooling, etc.) are addressed later in the procedure. This step was separated from them and moved earlier in the procedure (here) because of the potential time critical concern.

REFERENCES:
PIP M98-383
Calc MCC 1211.00-33.0006

From AP-07 (Loss of Electrical Power)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19. **Check the following DC pumps start as required:**	
__ a. Check Unit 1 6900V busses - AT ZERO VOLTS.	__ a. Observe Caution prior to Step 20 and GO TO Step 20.
__ b. Main Turbine "EMERG BRG OIL PUMP".	__ b. Start pump.
__ c. "DC B/U VAP EXTRACTOR".	__ c. Start vapor extractor.
__ d. "A CF PUMP TURB EBOP".	__ d. Start pump.
__ e. "B CF PUMP TURB EBOP".	__ e. Start pump.
__ f. Check OAC - AVAILABLE.	f. Perform the following: __ 1) Dispatch operator to ensure Unit 1 "AIR SIDE BACKUP" pump running (Unit 1 Turbine Bldg, 760, 1F-23). __ 2) Observe Caution prior to Step 20 and GO TO Step 20.
__ g. Check computer point M1D0581 (U1 Gen Air Side Seal Oil Backup Pump) - ON.	__ g. Dispatch operator to ensure Unit 1 "AIR SIDE BACKUP" pump running (Unit 1 Turbine Bldg, 760, 1F-23).

CAUTION **If any battery charger has lost power, then restarting the charger(s) in Enclosure 7 (DC Bus Alignment) is time critical.**

__ 20. **Have available licensed operator initiate Enclosure 7 (DC Bus Alignment) within 30 minutes of B/O.**

From AP-07 Bkgd Doc (Loss of Electrical Power) Pg 14

UNIT 1 CASE I STEP 19:
UNIT 2 CASE I STEP 17:

PURPOSE:

Ensure the major turbine building equipment (Main Turbine, CF Pumps, etc.) receive emergency DC powered lubrication following a loss of offsite AC power.

DISCUSSION:

This equipment may remain rotating for a period of time following the loss of offsite AC power. While rotating, lubrication is still required to prevent damage. Therefore, direction is provided to ensure the DC powered lubrication pumps are running to supply this lubrication. There is also a DC powered pump to ensure seal oil pressure to prevent a loss of generator hydrogen.

REFERENCES:

UNIT 1 CASE I STEP 20 CAUTION:
UNIT 2 CASE I STEP 18 CAUTION:

PURPOSE:

Cue the Operator the following step should have sufficient focus to ensure completion to meet the time critical nature of the step.

DISCUSSION:

The time critical nature of the step is: If power supply is lost to an essential battery charger (LOOP with failure of 1 D/G), it must be swapped to the other unit within an hour (MCC-1381.05-000-0220, 125VDC Vital Battery and Battery Charger Calculation).

The following step cues the operator to "start" the enclosure within 30 minutes. There is additional time critical assumptions made once the enclosure is initiated, so this caution ensures the operator is aware of these requirements.

REFERENCES:

Parent Question:

1 Pt Both Units are operating at 100% RTP.

- Loss of Offsite Power occurred on Unit 1
- Both DGs started and loaded as designed
- 1A D/G subsequently trips on overspeed
- At Step 17 of ES 0.1 (Reactor Trip Response), the decision is made to implement AP-07 (Loss of Electrical Power)

Which ONE (1) of the following correctly describes the Time Critical local operator actions associated with AP-07?

A. Implement Enc. 7 (DC Bus Alignment) to realign Battery Charger EVCA to Unit 2 within one hour.

B. Implement Enc. 7 (DC Bus Alignment) to realign Battery EVCA to Battery Charger EVCS within one hour.

C. Implement Generic Enc. 13 (VC and VA System Operation) to restart the Train A VC/YC Chiller within 37.5 minutes.

D. Implement Generic Enc. 13 (VC and VA System Operation) to swap Train A VC/YC Chiller power and water to Unit 2 and restart chiller within 1 hour and 15 minutes.

Proposed Answer: **A**

KA	KA_desc
SYS015 A2.01	Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5)□Power supply loss or erratic operation .....

Given the following conditions on Unit 1:

- The unit is currently at 80% RTP with a power increase in progress
- Power Range channel N-43 fails due to a faulty power supply
- N-43 has been removed from service in accordance with AP-16 (Malfunction of Nuclear Instrumentation)
- N-43 will be repaired in approximately 6 hours

1. In accordance with Tech Spec 3.2.4 (QPTR), Quadrant Power Tilt Ratios shall be determined by _____.
2. Quadrant Power Tilt limits prevent exceeding _____ power distribution design limits.

Which ONE (1) of the following completes the statements above?

A. 1. calculation using the remaining three Power Range channels OR movable incore detectors
   2. RADIAL

B. 1. using the movable incore detectors ONLY
   2. RADIAL

C. 1. calculation using the remaining three Power Range channels OR movable incore detectors
   2. AXIAL

D. 1. using the movable incore detectors ONLY
   2. AXIAL

General Discussion

With a power range channel out of service and power greater than 75% RTP, QPTR shall be determined by performing SR 3.2.4.2 (using incore detectors). If power was less than 75% RTP, QPTR is determined by calculation using the remaining three power range channels (SR 3.2.4.1). However, surveillance SR 3.2.4.1 allows performance of SR 3.2.4.2. (using incore detectors) in place of 3.2.4.1.

In accordance with Tech Spec 3.2.4 Basis:

"The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses."

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible because this would be correct if power was less than 75% RTP.

Part 2 is correct.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible because this would be correct if power was less than 75% RTP.

Part 2 is plausible if the applicant has the misconception that Quadrant Power Tilt limits prevent exceeding RADIAL distribution limits and AFD limits prevent exceeding AXIAL distribution limits.

Answer D Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is correct.

Part 2 is plausible if the applicant has the misconception that Quadrant Power Tilt limits prevent exceeding RADIAL distribution limits and AFD limits prevent exceeding AXIAL distribution limits.

Basis for meeting the KA

The KA is matched because a power supply failure for an NIS channel has occurred and the applicant must determine the impact on Quadrant Power Tilt determination in accordance with Tech Spec requirements.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the "Clarification Guidance for SRO-only Questions (Rev 1 dated 03/11/2010)" under the Screening Criteria for question linked to 10CFR55.43(b)(2) (Tech Specs):

1) It can NOT be answered solely by knowing < 1 hour Tech Specs
2) It can NOT be answered solely by knowing the LCO/TRM information listed "above-the-line"
3) It can NOT be answered by knowing the Tech Spec Safety Limits or their bases
4) It requires the applicant to have specific knowledge of surviellance requirements (SR 3.2.4.1 & 3.2.4.3) which are "below the line" and are greater than 1 hour surveillances. The applicant must also have knowledge of information contained in the Tech Spec 3.2.4 Basis Document. Specifically, the reason for having Quadrant Power Tilt limits (i.e. to prevent exceeding RADIAL power distribution limits) is contained in the Tech Spec basis document and in the surveillance requirements.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References

Learning Objective:
1) IC-ENB #19

References:
1) Technical Specification 3.2.4 QPTR

Student References Provided

2010 MNS SRO NRC Examination QUESTION 81

2581

B

2) Technical Specification 3.2.4 Basis

KA	KA_desc
SYS015 A2.01	Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5)□Power supply loss or erratic operation .....

401-9 Comments:	Remarks/Status

Question 81 References:

From Tech Spec 3.2.4 QPTR:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTES---------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER <75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. ---------- Verify QPTR is within limit by calculation.	7 days AND Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable
SR 3.2.4.2	----------NOTES---------- Only required to be performed if input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ 75% RTP. ---------- Verify QPTR is within limit using the movable incore detectors.	12 hours

From Tech Spec 3.2.4 Basis:

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), there must be a high level of probability that the peak cladding temperature does not exceed 2200°F (Ref. 1);

b. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}(X,Y)$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

BASES

ACTIONS (continued)

reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the more restrictive of the power level limit determined by Required Action A.1 or A.2 is exceeded. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the more restrictive limit of Required Action A.1 or A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.7 is modified by a Note that states that the peaking factor surveillances must be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.6). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

If Required Actions A.1 through A.7 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to ≤50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1 if more than one input from Power Range Neutron Flux channels are inoperable.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

BASES

SURVEILLANCE REQUIREMENTS (continued)

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is required only when the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore tilt. Therefore, incore tilt can be used to confirm that QPTR is within limits.

With one or more NIS channel inputs to QPTR inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

KA	KA_desc
SYS041	SYS041 GENERIC Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)
2.4.11	

Given the following conditions on Unit 1:

- The unit is operating at 10^{-8} AMPS taking critical data
- One condenser steam dump fails open
- Crew is performing AP-01 (Steam Leak)
- Pressurizer level is stable
- NC system temperature is 553°F and decreasing slowly

1. Based on the conditions above, to isolate the steam leak AP-01 will direct the crew to _____.

2. Isolating the steam leak is one of the design basis considerations for ensuring _____ per Tech Spec 3.4.2, RCS Minimum Temperature for Criticality Basis.

Which ONE (1) of the following completes the statements above?

A. 1. trip the Reactor and close the MSIVs
   2. the reactor remains subcritical in the event of a reactor trip

B. 1. take "A" and "B" "STEAM DUMP INTLK BYP" switches to "OFF/RESET"
   2. the reactor remains subcritical in the event of a reactor trip

C. 1. trip the Reactor and close the MSIVs
   2. proper indication and response of the excore detectors

D. 1. take "A" and "B" "STEAM DUMP INTLK BYP" switches to "OFF/RESET"
   2. proper indication and response of the excore detectors

General Discussion

In the scenario given, a main condenser steam dump has failed open during a start up with the unit in Mode 2 holding power at 10-8 amps. The crew has implemented AP-01 for a steam leak. Step 13 of AP-01 directs the crew to "Check condenser dump valves CLOSED" This would not be true so the RNO for this step directs the operators to select "OFF RESET" on Steam Dump Intk Bypass Channel A and B.

One of the major concerns of the CRS in this situation would be maintaining RCS temperature above the minimum temperature for criticality. The question solicits this basis which includes the consideration that excore NI would be adversely affected and if temperature were allowed to fall below this value, proper indication and required protective actions provided would not be assured.

Answer A Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because this would be the correct action if pressurizer level was decreasing (with maximum charging flow) or if NC system temperature was less than 551°F and decreasing.

Part 2 is plausible because it would be reasonable for the applicant to confuse the basis for Minimum temp for criticality with the basis for minimum temperature stated in many of our procedures which requires additional boration to prevent return to criticality.

Answer B Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is correct and therefore plausible.

Part 2 is plausible because it would be reasonable for the applicant to confuse the basis for Minimum temp for criticality with the basis for minimum temperature stated in many of our procedures which requires additional boration to prevent return to criticality.

Answer C Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because this would be the correct action if pressurizer level was decreasing (with maximum charging flow) or NC system temperature was less than 551°F and decreasing.

Part 2 is correct and therefore plausible.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

For this sencario, the applicant is given a malfunction of the Steam Dump System and is asked to demonstrate a knowledge of abnormal procedure actions related to operation of the steam dump controls to mitigate the consequences of the event. Therefore, the K/A is matched.

Basis for Hi Cog

This question is a higher cognitive level question because it requires the applicant to evaluate the plant conditions and determine the correct procedural actions based on the plant conditions.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge. Either of these will stop the steam leak. However, plant conditions dictate the procedural flowpath requirements which will direct the crew to take the Steam Dump INTLK BYP switch to OFF/RESET versus tripping the Reactor and closing the MSIVs.
2) The question can NOT be answered by knowing immediate operator actions. Neither of the actions described are immediate actions.
3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs. These are detailed procedure steps from AP-01.
4) The question can NOT be answered solely by knowing the purpose, overal sequence of events, or overall mitigative strategy of the procedure. This is detailed knowledge of procedure content related to knowing plant conditions that would require tripping the Reactor and closing the MSIVs as opposed to placing the steam dumps to OFF.
5) The question requires the applicant to analyze plant conditions and determine which section of the AP should be performed. Therefore, it is SRO knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	BANK Q 491

2010 MNS SRO NRC Examination QUESTION 82

2582

D

Development References	Student References Provided
'-01 (Rev 16) page (7 of 42) .S 3.4.2 Basis	

KA	KA_desc
SYS041	SYS041 GENERIC□Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)
2.4.11	

401-9 Comments:	Remarks/Status

Question 82 References:

From AP-01:

MNS AP/1/A/5500/01 UNIT 1	STEAM LEAK	PAGE NO. 3 of 42 Rev. 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. **Operator Actions**	
__ 1. **Monitor Foldout page.**	
2. **Reduce turbine load to maintain the following:**	
__ • Excore NI's - LESS THAN OR EQUAL TO 100%	
__ • NC Loop D/T's - LESS THAN 60°F D/T	
__ • T-Avg - AT T-REF.	
__ 3. **Check containment entry - IN PROGRESS.**	__ **GO TO Step 5.**
__ 4. **Check steam leak - KNOWN TO BE OUTSIDE CONTAINMENT.**	**IF conditions warrant, THEN evacuate containment as follows:**
	__ a. Announce "All personnel evacuate Unit 1 containment".
	__ b. Actuate the containment evacuation alarm.
	__ c. REFER TO RP/0/A/5700/011 (Conducting a Site Assembly, Site Evacuation, or Containment Evacuation) as time allows.
__ 5. **Check Pzr pressure prior to event - GREATER THAN P-11 (1955 PSIG).**	__ **IF AT ANY TIME an S/I occurs due to steam leak, THEN GO TO Enclosure 2 (S/I Actions For Steam Break In Modes 3 and 4).**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 6. **Check Pzr level - STABLE OR GOING UP.**	Perform the following as required to maintain level: __ a. Maintain charging flow less than 200 GPM at all times in subsequent steps. __ b. Ensure 1NV-238 (Charging Line Flow Control) OPENING. __ c. OPEN 1NV-241 (U1 Seal Water Inj Flow Control) while maintaining NC pump seal flow greater than 6 GPM. __ d. Reduce or isolate letdown. __ e. Start acditional NV pump. __ f. IF Pzr level going down with maximum charging flow, THEN GO TO Step 9.
__ 7. **IF AT ANY TIME while in this procedure Pzr level cannot be maintained stable, THEN RETURN TO Step 6.**	
__ 8. **GO TO Step 12.**	

Potential alternate flowpath.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Check unit status:**	**Perform the following:**
__ • Any shutdown or control rod - WITHDRAWN	
__ • Pzr pressure prior to event - GREATER THAN P-11 (1955 PSIG).	
Potential alternate flowpath.	
	NOTE If reactor trip breakers are closed and T-Avg is less than 553°F, a feedwater isolation will occur in the next step.
	__ a. Open reactor trip breakers.
	__ b. CLOSE all MSIVs using individual valve pushbuttons.
	c. **IF** EP/1/A/5000/ES-0.1 (Reactor Trip Response) has been implemented, **THEN** perform the following:
	1) THROTTLE S/G feed flow to:
	__ • Minimize cooldown
	__ • Maintain at least one S/G N/R level greater than 11%.
	__ 2) **GO TO** Step 11.
	d. **IF** feedwater isolation has occurred **AND** S/G levels going down in uncontrolled manner, **THEN** perform the following:
	__ 1) Start CA pump(s).
	__ 2) **WHEN** desired to feed S/Gs with CM/CF, **THEN REFER TO** AP/1/A/5500/06 (S/G Feedwater Malfunction).
	e. THROTTLE S/G feed flow to:
	__ • Minimize cooldown
	__ • Maintain at least one S/G N/R level greater than 11%.
	__ f. **GO TO** Step 11.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. Trip reactor and close MSIVs as follows:	
__ a. Trip reactor.	
__ b. CLOSE all MSIVs using individual valve pushbuttons.	
__ c. Continue with this AP as time allows.	
__ d. GO TO EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).	
11. **IF AT ANY TIME Pzr level goes below 4% AND cannot be restored using normal charging, THEN perform the following:**	
__ a. Ensure reactor tripped.	
__ b. **WHEN** reactor tripped **OR** auto S/I setpoint reached, **THEN** ensure S/I initiated.	
__ c. Check Pzr pressure prior to event - GREATER THAN P-11 (1955 PSIG).	__ c. GO TO Enclosure 2 (S/I Actions For Steam Break In Modes 3 and 4).
__ d. GO TO EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).	
__ 12. **Announce occurrence on paging system.**	

Potential alternate flowpath.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. **Identify and isolate leak on Unit 1 as follows:**	
__ a. Check SM PORVs - CLOSED.	a. **IF** S/G pressure is less than 1092 PSIG, **THEN** perform the following: __ 1) CLOSE affected S/G SM PORV manual loader. 2) **IF** SM PORV is still open, **THEN** perform the following: __ a) CLOSE SM PORV isolation valve. __ b) **IF** SM PORV isolation valve still open, **THEN** dispatch operator to CLOSE SM PORV isolation valve.
__ b. Check condenser dump valves - CLOSED.	b. **IF** steam dumps required to be closed, **THEN** perform the following: 1) Select "OFF RESET" on the following switches: __ • "STEAM DUMP INTLK BYPASS CHANNEL A" __ • "STEAM DUMP INTLK BYPASS CHANNEL B". __ 2) **IF** valve will not close, **THEN** dispatch operator to CLOSE condenser dump valve isolation valve. 3) **WHEN** leaking condenser dump valve is isolated **OR** repaired, **THEN** return the following switches to "ON": __ • "STEAM DUMP INTLK BYPASS CHANNEL A" __ • "STEAM DUMP INTLK BYPASS CHANNEL B".

From TS Basis for 3.4.2 (RCS Minimum Temperature for Criticality):

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The applicant could misinterpret this passage to imply a recriticallity concern

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

: the correct answer
in the question.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

APPLICABLE SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES (continued)

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 557°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

The special test exception of LCO 3.1.8, "PHYSICS TESTS Exceptions," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

Parent Question:

Initial conditions:

- Unit 1 is operating at 10-8 amps taking critical data
- One atmospheric steam dump fails opens
- Crew is performing AP-01 (Steam Leak)

What action is taken per AP-01 to attempt to close the dump valve, and what design bases consideration (per Tech Spec 3.4.2, RCS Minimum Temperature for Criticality) is assured if this action is successful?

A. Take "A" and "B" "STEAM DUMP INTLK BYP" switches to "OFF/RESET"
Steam generators are above their nil ductility reference temperature.

B. Dispatch operator to close the atmospheric dump valve isolation locally.
MTC will be in the range of slightly positive to negative.

C. Take "A" and "B" "STEAM DUMP INTLK BYP" switches to "OFF/RESET"
The pressurizer is within its normal startup and operating range.

D. Dispatch operator to close the atmospheric dump valve isolation locally.
Proper indication and response of the excore detectors when the reactor is critical.

KA	KA_desc
SYS002 A2.02	Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5)□Loss of coolant pressure .....

Given the following conditions on Unit 1:

- The unit is initially in MODE 3 with SD Banks withdrawn and NC System at full temperature and pressure
- 1NC-32B (PZR PORV) fails open
- AP-11 (Pressurizer Pressure Anomalies) has been implemented
- The PORV isolation valve for 1NC-32B will not close
- An RO is directed to trip the Reactor and initiate Safety Injection
- When attempted, both Reactor Trip breakers will not open

Which ONE (1) of the following describes the impact of the conditions above on the requirements for initiation of Safety Injection AND subsequent procedural transition?

A. The crew shall wait until the Reactor is tripped and then initiate Safety Injection, even if the low pressure Safety Injection setpoint is exceeded. The crew will then transition to AP-34 (Shutdown LOCA).

B. The crew shall wait until the Reactor is tripped and then initiate Safety Injection, even if the low pressure Safety Injection setpoint is exceeded. The crew will then transition to E-0 (Reactor Trip or Safety Injection).

C. The crew shall wait until the Reactor is tripped <u>OR</u> the low pressure Safety Injection setpoint is reached to initiate Safety Injection. The crew will then transition to AP-34 (Shutdown LOCA).

D. The crew shall wait until the Reactor is tripped <u>OR</u> the low pressure Safety Injection setpoint is reached to initiate Safety Injection. The crew will then transition to E-0 (Reactor Trip or Safety Injection).

General Discussion

In the scenario given with this question, the NCS has experienced a loss of pressure complicated by an ATWS (Failure of the RTB's to open from the C/R) Normally when approaching a ESF setpoint in a uncontrolled manor, the crew is expected to initiate the action prior to reaching the associated setpoint. In the case of an ATWS, the early initiation of SI would result in a FWI which could result in an extreme challenge to reactor safety (ATWS loss of Feedwater). In accordance with AP-11 Step 5 RNO, the crew should wait until the Reactor is tripped OR the SI setpoint is reached to ensure that SI is initiated.
The unit is in Mode 3 and the CLAs are not isolated so the correct transition is to go to E-0 (Reactor Trip or Safety Injection). If the unit was in Mode 3 with the CLAs isolated, the correct transition would be to AP-34 (Shutdown LOCA).

Answer A Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because initiating SI prior to the Reactor being tripped creates a worst case scenario ATWS event (i.e. ATWS with loss of feedwater). Therefore, it is reasonable for the applicant to conclude that SI should not be initiated until after the Reactor is tripped regardless of whether the SI setpoint is reached.

Part 2 is plausible because the unit is in Mode 3 and the appropriate transition would be to go to AP-34 if the CLAs were isolated.

Answer B Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because initiating SI prior to the Reactor being tripped creates a worst case scenario ATWS event (i.e. ATWS with loss of feedwater). Therefore, it is reasonable for the applicant to conclude that SI should not be initiated until after the Reactor is tripped regardless of whether the SI setpoint is reached.

Part 2 is correct and therefore plausible.

Answer C Discussion

INCORRECT: See explanation above.

LAUSIBLE: Part 1 is correct and therefore plausible.

Part 2 is plausible because the unit is in Mode 3 and the appropriate transition would be to go to AP-34 if the CLAs were isolated.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

The KA is matched because a loss of coolant pressure is occurring and the applicant must predict the impact of plant conditions on the procedural actions required to mitigate the event. The "predicting the impact part" of the KA is met because the applicant must determine how the procedural steps are different from the normal procedural flowpath based on a change in conditions (i.e. failure of the Reactor Trip breakers to open). The loss of pressure is responsible for the need to initiate SI.

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. The applicant must first analyze the given conditions and determine that the unit is in a Mode 3 condition where the CLAs could not be isolated. The applicant must then recall from memory that the correct transition in Mode 3 with the CLAs NOT isolated would be to go to E-0 (Reactor Trip or Safety Injection) as opposed to AP-34 (Shutdown LOCA). The applicant must also evaluate the impact of a ATWS complicated by a loss of NCS pressure resulting in the need to initiate SI. This represents an analysis of the situation to determine that the "normal" expectations of a crew which approaching an ESF setpoint in an uncontrolled manor would not apply in the situation.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) This question can NOT be answered by knowing systems knowledge alone. This is strict procedure knowledge. This is not covered during systems training or discussed in a systems lesson plan.
2) This question can NOT be answered by knowing immediate operator actions. The immediate actions from AP-11 address attempting to isolate the stuck open Pressurizer PORV. However, the actions to be taken by the crew as pressure continues to decrease are not part of the immediate actions.
3) This question can NOT be answered by knowing the entry conditions for AOPs. The steps to be taken by the crew are not based on the entry conditions provided.
4) This question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of the AOPs. The question is based on knowledge of specific procedure content.
5) The question requires the applicant to have in-depth knowledge of specific steps wtihin AP-11. Specifically, it requires the applicant to recall that the Step 5 RNO directs initiating SI only after the Reactor is tripped or the SI setpoint is reached. Additionally, the applicant must recall that

the same step directs transition to E-0 if the unit is in Mode 3 or above with the CLAs not isolated. Therefore, this is SRO level knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References	Student References Provided
Learning Objectives: 1) N/A References: 1) AP-11, Pressurizer Pressure Anomalies	

KA	KA_desc
SYS002	Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5)□Loss of coolant pressure .....
A2.02	

401-9 Comments:	Remarks/Status

Question 83 References:

From AP-11 (Pressurizer Pressure Anomalies):

MNS AP/1/A/5500/11 UNIT 1	PRESSURIZER PRESSURE ANOMALIES	PAGE NO. 2 of 9 Rev. 10

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

B. Symptoms

- Pzr pressure channel failed
- Pzr pressure going down in an uncontrolled manner
- Pzr pressure going up in an uncontrolled manner
- Any Pzr PORV or spray valve failed open
- "PZR PORV DISCH HI TEMP" alarm
- "PRT HI TEMP" alarm.

C. Operator Actions

__ 1. Check actual Pzr pressure - HAS GONE DOWN.	__ GO TO Step 17.
__ 2. Check all Pzr pressure channels - INDICATING THE SAME.	__ IF either controlling channel is malfunctioning, THEN place "PZR PRESS CNTRL SELECT" switch to backup channel.
__ 3. Check Pzr PORVs - CLOSED.	Perform the following: __ a. Close PORVs. __ b. IF PORV will not close, THEN close PORV isolation valve.
__ 4. Check Pzr spray valves - CLOSED.	Perform the following: __ a. Close Pzr spray valve(s). __ b. IF AT ANY TIME a reactor trip occurs AND spray valve still open, THEN stop 1A and 1B NC pumps.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 5. Check Pzr PORVs - CLOSED.	**Perform the following:** a. **IF** associated PORV isolation valve will not close **AND** pressure going down rapidly, **THEN**: 1) **IF** in Mode 3 or above, prior to CLA isolation, **THEN**: __ a) Trip reactor. __ b) **WHEN** reactor tripped **OR** auto S/I setpoint reached, **THEN** ensure S/I initiated. __ c) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). __ 2) **IF** in Mode 3 after CLA isolation or in Mode 4, **THEN GO TO** AP/1/A/5500/34 (Shutdown LOCA). b. Close associated PORV inlet drain valve as follows: __ • **IF** 1NC-32B (PZR PORV) failed, **THEN** close 1NC-271 (PZR PORV Drn Isol For 1NC-32B). __ • **IF** 1NC-34A (PZR PORV) failed, **THEN** close 1NC-270 (PZR PORV Drn Isol For 1NC-34A). __ • **IF** 1NC-36B (PZR PORV) failed, **THEN** close 1NC-269 (PZR PORV Drn Isol For 1NC-36B).

Distracter Plausibility

KA	KA_desc
APE015/017 AA2.02	Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): (CFR 43.5 / 45.13)□Abnormalities in RCP air vent flow paths and/or oil cooling system . . . . .

Given the following conditions on Unit 1:

- The unit is operating at 100% RTP
- 1C NCP Oil Reservoir Level alarm is received on the OAC
- Oil level indication on the OAC is -2.0 inches
- 1C NC pump motor bearing temperature is 200°F
- AP-08 (Malfunction of NC Pump) Case II (NC Pump Motor or Motor Bearing Malfunction) has been implemented

Which ONE (1) of the following describes the ACTIONS to be directed by the CRS in accordance with AP-08 and the HIGHEST POWER allowed at which the NCP can be stopped?

A. Trip the Reactor, verify reactor power less than 10%, then stop the 1C NCP.

B. Trip the Reactor, verify reactor power less than 5%, then stop the 1C NCP.

C. Perform a unit shutdown to MODE 3 using AP-04 (Rapid Downpower), then stop the 1C NCP.

D. Reduce the reactor power to < 48% using AP-04 (Rapid Downpower), then stop the 1C NCP.

General Discussion

In accordance with AP-08, if an NC pump must be stopped reactor must be tripped if the unit is operating in Mode 1 or 2 and the NC can not be stopped until power is less than 5%.

If the NC pump trip criteria has not yet been exceeded but it is determined that the NCP still needs to be stopped, AP-08 directs performing a unit shutdown in accordance with OP/1/A/6100/003 (Controlling Procedure for Unit Operation) or AP-04 (Rapid Downpower). The NCP is then stopped after all rods are inserted and the Reactor Trip breakers are open.

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible because tripping the Reactor first is the correct action. Stopping the NCP when power is less than 10% is plausible since the NC low flow trips are defeated when less than 10% power (P-10).

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible because it would be the correct action if the applicant concluded that the NCP needed to be stopped but did not need to be stopped immediately.

Answer D Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant concludes that the NCP needs to be stopped but does not need to be stopped immediately. Reducing reactor power to less than 48% would allow the NCP to be stopped without causing a reactor trip (less than the single loop low flow trip setpoint). Then the unit shutdown to MODE 3 would be continued since power operation is not allowed without all NCPs in service.

Basis for meeting the KA

he KA is matched because an malfunction of the oil cooling system has occurred and the applicant must determine the appropriate AP-08 actions based on plant conditions.

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. The applicant must first analyze the given plant conditions and determine that the 1C NCP must be stopped immediately. The applicant must then recall from memory the AP-08 actions required for stopping the NCP.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered by knowing systems knowledge. This is detail procedure content from AP-08.
2) The question can NOT be answered by knowing immediate Operator actions. None of the actions in the correct answer or in the distracters are immediate actions.
3) The question can NOT be answered by knowing entry conditions for the AP.
4) The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of AP-08.
5) The question require the applicant to assess plant conditions and determine appropriate actions based on detailed knowledge of procedure content. Specifically, the applicant must determine that the NC pump needs to be stopped immediately which requires the Reactor to be tripped first and power verified less than 5% before the NCP can be stopped. Therefore, this is SRO level knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References

Learning Objectives:
) N/A

References:
1) AP-08, Malfunction of NC Pump

Student References Provided

2010 MNS SRO NRC Examination QUESTION 84

2584

B

KA	KA_desc
APE015/017 AA2.02	Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): (CFR 43.5 / 45.13)□Abnormalities in RCP air vent flow paths and/or oil cooling system . . . . .

401-9 Comments:	Remarks/Status

Question 84 References:

From AP-08:

MNS AP/1/A/5500/08 UNIT 1	MALFUNCTION OF NC PUMP Case II NC Pump Motor or Motor Bearing Malfunction	PAGE NO. 14 of 24 Rev. 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B. Symptoms • **NC pump stator winding temperature going up** • **NC pump motor bearing temperatures going up** • **NC pump upper/lower oil reservoir level computer alarm.**	
C. Operator Actions	
__ 1. **Check abnormal NC pump parameter - KNOWN TO BE VALID.**	__ **GO TO Enclosure 1 (Validation of NC Pump Parameters).**
2. **Check NC pump parameters within operating limits:**	__ **IF trip criteria valid, THEN GO TO Step 5.**
__ • All NC pump stator winding temperatures - LESS THAN 311°F	
__ • All NC pump motor bearing temperatures - LESS THAN 195°F	
__ • All NC pump oil reservoir level computer points - INDICATING BETWEEN (-)1.25 AND (+)1.25.	
__ 3. **IF AT ANY TIME any operating limit in Step 2 exceeded, THEN GO TO Step 5.**	
__ 4. **GO TO Step 6.**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Stop affected NC pump as follows:**	
a. **IF** A or B NC pump is the affected pump, **THEN** CLOSE associated spray valve:	
__ • 1NC-27C (A NC Loop PZR Spray Control)	
__ • 1NC-29C (B NC Loop PZR Spray Control).	
__ b. Check unit status - IN MODE 1 OR 2.	b. Perform the following:
	__ 1) Stop the affected pump.
	2) **IF** all NC pumps are off, **THEN** perform the following:
	__ a) Secure any boron dilution in progress.
	__ b) **IF** in Mode 3, **THEN** immediately open Reactor Trip Breakers.
	__ c) **IF** the step above results in rods dropping **AND** Pzr pressure is above P-11, **THEN GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
	__ 3) **GO TO** Step 6.
__ c. Trip reactor.	
__ d. **WHEN** reactor power less than 5%, **THEN** stop affected NC pump.	
__ e. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).	
__ 6. **Announce occurrence on paging system.**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. Monitor the following NC pump parameters: __ • Stator winding temperatures (OAC) - STABLE OR GOING DOWN __ • Motor bearing temperatures (OAC) - STABLE OR GOING DOWN __ • Vibration - NORMAL __ • Oil reservoir levels (OAC) - STABLE.	**Perform the following:** __ a. Evaluate need to stop affected NC pump based on rate of change of abnormal NC pump parameter(s). b. IF NC pump needs to be stopped, THEN perform the following: 1) Begin unit shutdown to Mode 3 PER one of the following: __ • OP/1/A/6100/003 (Controlling Procedure For Unit Operation), Enclosure 4.2 (Power Reduction) OR __ • AP/1/A/5500/04 (Rapid Downpower). 2) **WHEN** in Mode 3, 4, or 5, **THEN** perform the following: a) **IF all** NC pumps need to be stopped, **THEN** perform the following: __ (1) Secure boron dilution. __ (2) Do not continue until rods inserted and reactor trip breakers open. b) **IF** A or B NC pump is the affected pump, **THEN** CLOSE associated spray valve: __ • 1NC-27C (A NC Loop PZR Spray Control) __ • 1NC-29C (B NC Loop PZR Spray Control). __ c) Stop affected NC pump.
__ 11. **Check NC pumps - ANY RUNNING.**	__ **IF both ND pumps off AND no EP in effect, THEN REFER TO AP/1/A/5500/09 (Natural Circulation) as time allows.**

Distracter Plausibility

END

KA	KA_desc
APE022	APE022 GENERIC Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)
2.4.46	

Unit 1 is operating at 100% RTP when the following alarms are received:

- 1AD-7 / J1 (NC PUMP SEAL INJ LO FLOW)
- 1AD-7 / I2 (REGEN HX LETDN HI TEMP)
- 1AD-7 / G2 (CHARGING LINE ABNORMAL FLOW)

The crew has implemented AP-12 (Loss of Letdown, Charging, or Seal Injection).

1. Based on plant conditions indicated by the alarms above, what actions are directed by AP-12?

2. What actions are directed by AP-12 regarding the restoration of letdown during the subsequent recovery?

A.
1. FIRST close the Letdown Orifice Isolations (1NV-458A, 457A, 35A) and then close 1NV-1A, 2A (NC L/D Isol To Regen Hx).
2. Pressurize the Letdown system locally.

B.
1. Close 1NV-1A, 2A (NC L/D Isol To Regen Hx) which will auto-close the Letdown Orifice Isolations (1NV-458A, 457A, 35A).
2. Pressurize the Letdown system locally.

C.
1. FIRST close the Letdown Orifice Isolations (1NV-458A, 457A, 35A) and then close 1NV-1A, 2A (NC L/D Isol To Regen Hx).
2. Pressurize the Letdown system from the Control Room.

D.
1. Close 1NV-1A, 2A (NC L/D Isol To Regen Hx) which will auto-close the Letdown Orifice Isolations (1NV-458A, 457A, 35A).
2. Pressurize the Letdown system from the Control Room.

General Discussion

In accordance with AP-12, Loss of Letdown, Charging, or Seal Injection, the crew should first close the Letdown Orifice Isolations since they have indications that charging has been lost. Then because the Regen Hx Letdown Hi Temp alarm is in, they should close the isolations to the Regen Hx (NV-1A, 2A).

During subsequent recovery actions, the crew is procedurally directed to pressurize the letdown system from the control room since the Letdown Orifice Isolations were closed prior to NV-1A and 2A. Had NV-1A and 2A closed first, the crew would be required to pressurized the letdown system locally during restoration.

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is correct.

Part 2 is plausible if the applicant does not recall that the letdown line is pressurized locally when the Letdown Orfice Isolation Valves (1NV-458A, 457A, & 35A) close prior to 1NV-1A & 2A.

Answer B Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible if the applicant just remembers the step for closing 1NV-1A and 2A which states:

"IF AT ANY TIME, "REGEN HX LTDN HI TEMP" alarm (1AD7-I2) is LIT, THEN close the following valves:
1NV-1A
1NV-2A

Procedurally the Letdown Orfice Isolations should have already been closed because the alarms in combination provide positive indication that a loss of charging has occurred and the steps to close the orifice isolations come before the steps to close NV-1A and 2A in the RNO column.

If the applicant concludes that closing the Letdown Orifice Isolations first is the correct response then the Letdown Line would have to be pressurized locally in accordance with AP-12 making Part 2 correct.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible if the applicant just remembers the step for closing 1NV-1A and 2A which states:

"IF AT ANY TIME, "REGEN HX LTDN HI TEMP" alarm (1AD7-I2) is LIT, THEN close the following valves:
1NV-1A
1NV-2A

Procedurally the Letdown Orfice Isolations should have already been closed because the alarms in combination provide positive indication that a loss of charging has occurred and the steps to close the orifice isolations come before the steps to close NV-1A and 2A in the RNO column.

Part 2 is plausible if the applicant does not recall that the letdown line is pressurized locally when the Letdown Orfice Isolation Valves (1NV-458A, 457A, & 35A) close prior to 1NV-1A & 2A.

Basis for meeting the KA

The applicant must analyze the combination of alarms given in the stem of the question to determine the condition of the plant (i.e. in this case that a loss of charging has occurred). The applicant demonstrates that they have correctly identified plant conditions by selecting the correct actions from AP-12 for that plant condition. If the applicant choses the correct procedure actions based on their conclusions regarding plant conditions, they have demonstrated the ability to verify that the alarms are consitent with plant conditions. Therefore, the KA is matched.

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. The applicant must first diagnose the conditions given to determine what has caused the alarms. The applicant must then recall from memory the procedure requirements for isolating letdown and the requirements for recovering letdown.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) This question can NOT be answered by knowing systems knowledge alone. This requires tha applicant to analyze a given set of alarms and

determine what plant conditions could have caused that combination of alarms. The applicant must then determine what procedural actions from AP-12 are appropriate for plant conditions.
2) This question can NOT be answered by knowing immediate operator actions. AP-12 has no immediate actions.
3) This question can NOT be answered by knowing the entry conditions for AOPs. The alarms given are entry conditions for AP-12 However, the applicant is given that AP-12 has been entered and determine what actions from the procedure are appropriate based on the combination of alarms.
4) This question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of the AOPs.
5) This question DOES require the applicant to assess plant conditions (based on a combination of alarms) and determine from that assessment the appropriate steps from AP-12 to be taken. This requires the applicant to have detailed knowledge of specific procedure steps from AP-12.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References
Learning Objectives: 1) AP12002 References: 1) AP-12, Loss of Letdown, Charging, or Seal Injection

Student References Provided

KA	KA_desc
APE022	APE022 GENERIC□Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)
2.4.46	

401-9 Comments:	Remarks/Status

Question 85 References:

From AP-12 (Loss of Letdown, Charging, or Seal Injection):

MNS AP/1/A/5500/12 UNIT 1	LOSS OF LETDOWN, CHARGING OR SEAL INJECTION	PAGE NO. 3 of 43 Rev. 22

C. Operator Actions

1. **Check if charging is aligned to Regenerative Hx as follows:**

   ☐ Charging flow - GREATER THAN 20 GPM

   __ ☐ 1NV-241 (U1 Seal Water Inj Flow Control) - THROTTLED OPEN

   __ ☐ 1NV-244A (Charging Line Cont Outside Isol) - OPEN

   __ ☐ 1NV-245B (Charging Line Cont Outside Isol) - OPEN.

While the applicant is not given these specific indications, they should be able to determine from the combination of alarms that charging flow has been lost.

IF loss of charging through Regenerative Hx has occurred, THEN perform the following:

a. Ensure the the following valves are CLOSED:

   __ ☐ 1NV-458A (75 GPM L/D Orifice Outlet Cont Isol)

   __ ☐ 1NV-457A (45 GPM L/D Orifice Outlet Cont Isol)

   __ ☐ 1NV-35A (Variable L/D Orifice Outlet Cont Isol).

__ b. IF 1NV-2A (NC L/D Isol To Regen Hx) is closed, THEN GO TO Step 2.

c. IF AT ANY TIME "REGEN HX LETDN HI TEMP" alarm (1AD-7, -2) is lit, THEN CLOSE the following valves:

   __ ☐ 1NV-1A (NC L/D Isol To Regen Hx)

   __ ☐ 1NV-2A (NC L/D Isol To Regen Hx).

Letdown Recovery Actions from AP-12:

MNS AP/1/A/5500/12 UNIT 1	LOSS OF LETDOWN, CHARGING OR SEAL INJECTION	PAGE NO. 20 of 43 Rev. 22

42. **Check the following valves - OPEN:**	**Perform the following:**
__ ☐ 1NV-1A (NC L/D Isol To Regen Hx)	__ a. IF normal letdown known to be unavailable, THEN GO TO Step 49.
__ ☐ 1NV-2A (NC L/D Isol To Regen Hx).	__ b. Prior to opening 1NV-1A or 1NV-2A in subsequent step, ensure that all personnel are out of lower containment (potential water hammer event).
	__ c. Observe Caution prior to Step 44 and GO TO Step 44.

__ 43. **GO TO Step 48.**

CAUTION **Establishing normal letdown without local pressurization may cause some water hammer.**

Alternate Path

44. **Determine if conditions allow immediate restoration of normal letdown as follows:**	
__ a. Check both 1NV-1A (NC L/D Isol To Regen Hx) and 1NV-2A (NC L/D Isol To Regen Hx) - OPEN WITHIN THE LAST 60 MINUTES.	a. GO TO RNO for Step 44.d.
__ b. Check orifice isolation valves - AUTO CLOSED.	b. Perform the following:
	__ 1) IF orifice isolation valves were manually closed while both 1NV-1A and 1NV-2A were open, THEN GO TO Step 48.
	__ 2) GO TO RNO for Step 44.d.

44. (Continued)

c. Determine exact time each NV letdown valve went closed on the OAC by performing the following:

__ 1) Enter turn on code "ARCHIVE".

__ 2) Ensure OAC automatically populates "START TIME" and STOP TIME". (previous hour).

__ 3) Enter group name "AP12".

__ 4) Click "F3 - VIEW PID".

__ c. **IF** unable to determine time of closure, **THEN GO TO** RNO for Step 44.d.

__ d. Check if orifice isolation valves reached fully closed - PRIOR TO INV-1A OR INV-2A CLOSING.

d. Perform the following:

__ 1) **IF** excess letdown is in service, **THEN** observe Caution and Note prior to Step 46 and **GO TO** Step 46.

NOTE Establishing normal letdown requires local pressurization of letdown header. Since this action takes significant time, establishing excess letdown first may be desired.

__ 2) **IF AT ANY TIME** it is desired to establish excess letdown, **THEN GO TO** Step 49.

__ 3) Observe Caution and Note prior to Step 46 and **GO TO** Step 46.

__ 45. GO TO Step 48.

CAUTION **It is preferable to locally pressurize the letdown line prior to establishing letdown flow due to possible water hammer.**

NOTE If plant conditions require immediate restoration of normal letdown, OSM may waive the requirement to locally pressurize the letdown header.

__ 46. **IF normal letdown is required prior to locally pressurizing letdown header, THEN GO TO Step 18.**

If the applicant concludes that closing the Orifice Isolation Valves first is the correct response or does not recall when the letdown line should be pressurized locally (relative to the closure of the Letdown Isolations and the Letdown Orifice Isolations) they would end up at this step

47 Locally pressurize letdown header as follows:

__ a. Contact RP to evaluate containment conditions for entry into pipechase.

__ b. IF desired, THEN place Containment Aux Carbon Filter System in operation PER OP1/A/0400/015 (Containment Purge System), Enclosure 4.1 (Containment Aux Carbon Filter System Operation).

__ c. Dispatch operator to check letdown header pressure on local gauge 1NVPG5560 (Letdown Pressure) (Unit 1 reactor bldg. 725+3, pipechase, 218 degrees, on inner wall right of the ladder to C accumulator) - LESS THAN 1200 PSIG.

__ c. GO TO Step 48.

__ d. Have operator in containment OPEN 1NV 114 (Unit 1 Regenerative Hx Shell Side Flush & Vent Isol) (Unit 1 reactor bldg. 725+9, pipechase, 119 degrees, 2 ft from inner wall).

__ e. Do not continue until 1NV-114 is open.

__ f. Establish charging flow to Regenerative Hx by THROTTLING OPEN 1NV-241 (U1 Seal Water Inj Flow Control) until change in charging flow is indicated.

48 **Establish normal letdown as follows:**

__ a. Ensure 1NV-459 (U1 Variable L/D Orifice Outlet Flow Cntr) is CLOSED.

__ b. Place 1NV-124 (Letdown Pressure Control) in manual between 10-20% OPEN.

c. Check the following valves - OPEN:

__ ☐ 1NV-1A (NC L/D Isol To Regen Hx)

__ ☐ 1NV-2A (NC L/D Isol To Regen Hx).

__ GO TO Step 49.

This is the beginning of the sequence of steps to pressurize letdown from the Control Room. You end up here regardless of how long letdown was isolated provided 1NV-1A & 2A closed prior to the orifice isolation valves.

__ c. Ensure all personnel are out of lower containment prior to continuing.

CAUTION **A Pzr insurge will occur when charging flow is raised in next step. Letdown should be established without delay to limit the amount of insurge.**

d. Establish cooling to Regenerative Hx by performing the following concurrently:

__ ☐ Establish at least 55 GPM charging flow by THROTTLING OPEN 1NV-238 (Charging Line Flow Control) or raising PD pump speed.

__ ☐ THROTTLE 1NV-241 (U1 Seal Water Inj Flow Control) to establish approximately 8 GPM seal injection flow to each NC pump.

__ d. IF charging flow to Regenerative Hx cannot be established, THEN GO TO Step 49.

e. OPEN letdown line isolation valves as follows:

__ 1) OPEN 1NV-7B (Letdown Cont Outside Isol).

2) OPEN 1NV-1A (NC L/D Isol To Regen Hx).

__ 3) OPEN 1NV-2A (NC L/D Isol To Regen Hx).

__ 4) OPEN 1NV-35A (Variable L/D Orifice Outlet Cont Isol).

__ e. GO TO Step 49.

KA	KA_desc
APE027 AA2.10	Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: (CFR: 43.5 / 45.13)□PZR heater energized/de-energized condition .....

Given the following conditions on Unit 1:

- 1A and 1D Pressurizer heater group supply breakers open at 1100 on June 1 due to a lightning strike and cannot be reclosed
- A reactor startup is in progress with reactor power at 1% RTP
- Heater groups 1B, and 1C are available

Which ONE (1) of the following describes the required actions per Tech Spec 3.4.9, (Pressurizer)?

REFERENCE PROVIDED

A. Restore PZR heater group 1A ONLY to operable status.

B. Restore PZR heater group 1D ONLY to operable status.

C. Restore PZR heater group 1A AND 1D to operable status.

D. Be in MODE 4 in 13 hours in accordance with Tech Spec 3.0.3.

General Discussion

The 1A and 1D Pressurizer heater groups have been de-energized. The remaining two groups of heaters are capable of maintianing Pressure under normal conditions. However, the OPERABILITY requirement is based on two groups of heaters with a capacity of 150KW each with each group being capable of being supplied by off-site or emergency power. Since Pressurizer heater groups 1A and 1B are the only groups which can be supplied by emergency power, both groups are required to be OPERABLE in Modes 1, 2, and 3. Therefore, PZR heater group 1A must be returned to operable status. Group 1D is not required for TS operability.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant does not recall from the Tech Spec Basis which groups of heaters meet the operability requirements for Tech Specs. TS 3.4.9 requires two groups of heaters with a capacity of 150KW each with each group being capable of being supplied by off-site or emergency power. Since Pressurizer heater groups 1A and 1B are the only groups which can be supplied by emergency power, both groups are required to be OPERABLE in Modes 1, 2, and 3.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant does not recall from the Tech Spec Basis which groups of heaters meet the operability requirements for Tech Specs. TS 3.4.9 requires two groups of heaters with a capacity of 150KW each with each group being capable of being supplied by off-site or emergency power. Since Pressurizer heater groups 1A and 1B are the only groups which can be supplied by emergency power, both groups are required to be OPERABLE in Modes 1, 2, and 3.

Answer D Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant does not recall from the Tech Spec Basis which groups of heaters meet the operability requirements for Tech Specs. TS 3.4.9 requires two groups of heaters with a capacity of 150KW each with each group being capable of being supplied by off-site or emergency power. Since Pressurizer heater groups 1A and 1B are the only groups which can be supplied by emergency power, both groups are required to be OPERABLE in Modes 1, 2, and 3. If the applicant concludes that 1A and 1D heaters are the two groups of heaters required for operability, the appropriate action would be to enter TS 3.0.3 since the is no action statement for two of the required groups being inoperable. For example, if Pressurizer heater groups 1A and 1B were inoperable this would be a correct answer.

Basis for meeting the KA

Strict knowledge of Pressurizer heater energized/de-energized conditions is RO level knowledge. However, given a condition were a group of Pressurizer heaters is de-energized and asking applicant to determine if the TS operability requirements for Pressurizer heaters are met and having them apply Tech Specs to determine an appropriate action raises the question to the SRO level while achieving a match to the "interpret" portion of the KA.

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. It requires the applicant to recall from memory that the Tech Spec requirement for operable heaters only applies to those with emergency power supplies (i.e. Group 1A and 1B). The applicant must the correctly apply Tech Specs to determine the correct actions to be taken.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the "Clarification Guidance for SRO-only Questions (Rev 1 dated 03/11/2010)" under the Screening Criteria for question linked to 10CFR55.43(b)(2) (Tech Specs):

1) It can NOT be answered solely by knowing < 1 hour Tech Specs. There are no actions in TS 3.4.9 which are required to be completed in 1 hour or less.
2) It can NOT be answered solely by knowing the LCO/TRM information listed "above-the-line". This question requires the applicant to recall information from the Basic Document for TS 3.4.9 to correctly apply the specification.
3) It can NOT be answered by knowing the Tech Spec Safety Limits or their bases. All actions are associated with TS 3.4.9, Pressurizer.
4) It DOES require the applicant to apply required actions and have additional knowledge contained in the Tech Spec Basis (specifically what constitues Pressurizer heater operability) to be able to apply the specification correctly and arrive at the correct answer.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

2010 MNS SRO NRC Examination QUESTION 86

2586

A

Development References	Student References Provided
Learning Objectives: 1) PS-NC#24 References: 1) Tech Spec 3.4.9, Pressurizer 2) Tech Spec 3.4.9 Basis	Tech Spec 3.4.9, Pressurizer

KA	KA_desc
APE027	Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: (CFR: 43.5 / 45.13)□PZR heater energized/de-energized condition .....
AA2.10	

401-9 Comments:	Remarks/Status

Question 86 References:

From Tech Spec 3.4.9:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 92% (1600 ft^3); and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open. AND A.2 Be in MODE 4.	6 hours 12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

From Tech Spec 3.4.9 Basis:

BASES

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for pressurizer level to remain within the required range is consistent with the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the UFSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 3), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1600 cubic feet, which is equivalent to 92%, ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with safety analysis analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW, capable of being powered from either the offsite power source or the emergency power supply. Only heater groups A and B are capable of being powered from the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The amount needed to maintain pressure is dependent on the heat losses.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS

KA	KA_desc
EPE038 2.4.11	EPE038 GENERIC Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

Given the following conditions on Unit 1:

- NC system is in MODE 3
- S/G tube leakage occurs on "C" S/G
- AP-10 (NC System Leakage within the Capacity of Both NV Pumps), Case I (S/G Tube Leakage) has been implemented
- The crew has reached the step in AP-10 to perform a rapid cooldown

The target temperature for the cooldown is determined based on ___(1)___.

The basis for the cooldown to the selected target temperature is ___(2)___.

Which ONE (1) of the following completes the statements above?

A. 1. ruptured SG pressure
   2. to ensure the NC system is 20°F subcooled following the NC system depressurization

B. 1. lowest intact SG pressure
   2. to ensure the NC system is 20°F subcooled following the NC system depressurization

C. 1. ruptured SG pressure
   2. to ensure NC system temperature is below the saturation temperature for the ruptured SG PORV lift pressure

D. 1. lowest intact SG pressure
   2. to ensure NC system temperature is below the saturation temperature for the ruptured SG PORV lift pressure

General Discussion

In accordance with the AP-10 Basis Document for Case 1 Step 25:

The principal goal of the AP is to minimize and eventually stop primary-to-secondary leakage. This step is designed to determine the target temperature that will establish sufficient subcooling in the NC so that the primary system will remain subcooled after NC pressure is decreased in subsequent steps to stop primary-to-secondary leakage.

Since, in order to stop this leakage, the NC pressure must be decreased to a value equal to the affected steam generator pressure, the temperature at which this cooldown is terminated is dependent upon the affected steam generator pressure. A table is constructed for various affected steam generator pressures showing the fluid temperature corresponding to 20°F subcooling at each of these pressures, including allowances for subcooling uncertainties. For consistency with the EPs, the target temperature should be based on the core exit TCs. The 20°F subcooling is provided as operating margin to accommodate fluctuations in NC temperature, perturbations in affected steam generator pressure, interpolation between listed affected steam generator pressures, and overshoot during NC depressurization. S/G pressure ranges were specified as human factors enhancement.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible if the applicant does not recall which pressure is used to determine the target temperature as they might conclude that the ruptured SG would be at a higher pressure (due to the NC system inleakage) than the intact SGs and that it would be appropriate to chose the intact SG with the lowest pressure to determine the target temperature for the cooldown. This would yield the lowest target temperature and provide for the most subcooling.

Part 2 is correct.

Answer C Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is correct.

Part 2 is plausible because part of the strategy in AP-10 with regards to isolating the ruptured SG and conducting the cooldown is minimizing the possiblity of lifting the SG PORV on the ruptured SG. For example, during the isolation of the ruptured SG, the basis for closing the MSIV last is to minimize the time between when the SG is bottled up and commencement of the rapid cooldown to minimize the possibility of lifting the PORV. So, it is plausible for the applicant to conclude that the target temperature is selected based on establishing conditions which will not result in the PORV lifting.

Answer D Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible if the applicant does not recall which pressure is used to determine the target temperature as they might conclude that the ruptured SG would be at a higher pressure (due to the NC system inleakage) than the intact SGs and that it would be appropriate to chose the intact SG with the lowest pressure to determine the target temperature for the cooldown. This would yield the lowest target temperature and provide for the most subcooling.

Part 2 is plausible because part of the strategy in AP-10 with regards to isolating the ruptured SG and conducting the cooldown is minimizing the possiblity of lifting the SG PORV on the ruptured SG. For example, during the isolation of the ruptured SG, the basis for closing the MSIV last is to minimize the time between when the SG is bottled up and commencement of the rapid cooldown to minimize the possibility of lifting the PORV. So, it is plausible for the applicant to conclude that the target temperature is selected based on establishing conditions which will not result in the PORV lifting.

Basis for meeting the KA

The K/A is matched because the applicant must have detailed knowledge of steps from the abnormal procedure for dealing with SG Tube Leaks (AP-10, NC System Leakage Within the Capacity of Both NV Pumps - Case 1, Steam Generator Tube Leakage) and knowledge of the basis for steps from the AP.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered by knowing systems knowledge. This is knowledge of detailed procedure content.

2) The question can NOT be answered by knowing immediate Operator actions. There are no immediate actions in AP-10.
The question can NOT be answered by knowing entry conditions for the AP-10.
, The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of AP-10.
5) The question requires the applicant to have knowledge of detailed procedure content from AP-10 (specifically what parameter is used to determine the target temperature for the rapid cooldown) and the basis (from the AP-10 Background Document) for performing the cooldown. Therefore, this is SRO level knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References
Learning Objective: 1) N/A References: 1) AP-10 (NC System Leakage Within the Capacity of Both NV Pumps) Case I (Steam Generator Tube Leakage)

Student References Provided

KA	KA_desc
EPE038	EPE038 GENERIC Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)
2.4.11	

401-9 Comments:

Remarks/Status

Question 87 References:

From AP-10:

MNS AP/1/A/5500/10 UNIT 1	NC SYSTEM LEAKAGE WITHIN THE CAPACITY OF BOTH NV PUMPS Case I Steam Generator Tube Leakage	PAGE NO. 3 of 129 Rev. 22

C. Operator Actions

__ 1. **Check Pzr level - STABLE OR GOING UP.**

Perform the following as required to maintain level:

__ a. Maintain charging flow less than 200 GPM at all times in subsequent steps.

__ b. Ensure 1NV-238 (Charging Line Flow Control) OPENING.

__ c. OPEN 1NV-241 (U1 Seal Water Inj Flow Control) while maintaining NC pump seal flow greater than 6 GPM.

__ d. Reduce or isolate letdown.

__ e. Start additional NV pump.

f. **IF** CLAs are isolated, **AND** Pzr level is going down, **THEN** perform the following:

__ 1) **REFER TO** Enclosure 1 (Shutdown Mode SGTL Actions To Restore Pzr Level)

__ 2) **GO TO** Step 2.

This provides plausibility for distracter D. With the depressed PZR level at 5%, this would seem a reasonable course but conditions given in the stem do not support this path.

g. **IF** Pzr level cannot be maintained greater than 4%, **OR** Pzr level going down with maximum charging flow, **THEN** perform the following:

__ 1) **IF** 1B **OR** 1C S/G identified as ruptured, **THEN** immediately dispatch two operators to isolate TD CA pump steam supply from ruptured S/G **PER** Enclosure 3 (TD CA Pump Steam Supply Isolation).

__ 2) Trip reactor.

__ 3) **WHEN** reactor tripped **OR** auto S/I setpoint reached, **THEN** ensure S/I initiated.

__ 4) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)

From AP-10

7. **Check if unit shutdown or reactor trip required as follows:**

__ a. Check VCT makeup - IN PROGRESS.

__ b. Check VCT level - GOING UP.

__ c. Check S/G tube leak size - LESS THAN 90 GPM.

__ a. GO TO Step 7.c.

__ b. Assume S/G tube leak size in next step is greater than 90 GPM.

c. Perform the following:

1) Align NV pumps to FWST as follows:

a) OPEN the following valves:

__ • 1NV-221A (NV Pumps Suct From FWST)

__ • 1NV-222B (NV Pumps Suct From FWST).

b) CLOSE the following valves:

__ • 1NV-141A (VCT Outlet Isol)

__ • 1NV-142B (VCT Outlet Isol).

2) IF reactor trip breakers closed, THEN perform the following:

__ a) Start both MD CA pumps.

__ b) Trip reactor.

__ c) IF Pzr pressure was less than 1955 PSIG prior to this event, THEN GO TO Step 9.

__ d) GO TO EP/1/A/5000/E-0 (Reactor Trip or Safety Injection), while continuing with this EP as time allows beginning at Step 9.

__ 3) IF reactor trip breakers are open, THEN GO TO Step 9.

This is the correct answer C. Direction is given here to go to Step 9 which will keep the crew in Ap-10 for the recovery.

This direction provides plausibility for distracter B. If pressure was above P-11 (1955 psig), this would be the correct answer.

KA	KA_desc
APE058 AA2.02	Ability to determine and interpret the following as they apply to the Loss of DC Power: (CFR: 43.5 / 45.13)□125V dc bus voltage, low/critical low, alarm .....

. Given the following conditions on Unit 1:

- The unit was operating at 100% RTP when a total loss of onsite and offsite power occurred

1. In accordance with AP-15 (Loss of Vital or Aux Control Power), what is the MINIMUM voltage on the DC Vital busses which requires the Vital Batteries (EVCA, EVCB, EVCC, EVCD) to be removed from service?

2. After power is restored and the battery chargers are placed in service, in accordance with Tech Spec 3.8.4 (DC Sources – Operating), what is the MINIMUM voltage required for the Vital Batteries to be OPERABLE while on float charge?

A. 1. 110 volts
   2. 125 volts

B. 1. 105 volts
   2. 125 volts

C. 1. 110 volts
   2. 110 volts

D. 1. 105 volts
   2. 110 volts

General Discussion

In accordance with Tech Spec 3.8.4 Basis (DC Sources Operating):

"The minimum battery terminal voltage limit is greater than or equal to 125 V while on float charge as discussed in the UFSAR, Chapter 8 (Ref. 4). "

In accordance with AP-15 (Loss of Vital or Aux Control Power) the Battery EVCA Switch must be opened if Bus EVDA voltage decreases to 105 volts.

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible because because if the battery discharged to less than 110 volts surveillance SR 3.8.6.2 must be performed to verify that battery cell paramters are within limits.

Part 2 is correct.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Both parts are plausible because if the battery discharged to less than 110 volts surveillance SR 3.8.6.2 must be performed to verify that battery cell paramters are within limits.

Answer D Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is correct.

Part 2 is plausible because if the battery discharged to less than 110 volts surveillance SR 3.8.6.2 must be performed to verify that battery cell paramters are within limits.

Basis for meeting the KA

The KA is matched because the applicant must be familiar with the minimum ("low/critical low") voltage at which the vital battery must be separated from the vital battery bus.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the "Clarification Guidance for SRO-only Questions (Rev 1 dated 03/11/2010)" under the Screening Criteria for question linked to 10CFR55.43(b)(5) (Assessment and Selection of Procedures) and 10CFR55.43(b)(2) (Tech Specs):

Part 1:

1) The question can NOT be answered by knowing systems knowledge. The minumum voltage on the bus before the battery has to be removed from service is only addressed in AP-15. This voltage is NOT covered by the systems lesson plan or taught during systems training. Therefore, it is not systems knowledge.
2) The question can NOT be answered by knowing immediate Operator actions. There are no immediate actions associated with AP-15.
3) The question can NOT be answered by knowing entry conditions for the AP. The information tested does not constitue entry conditions for AP-15.
4) The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of AP-15.
5) The question requires the applicant to have knowledge of specific diagnositic steps within AP-15. Specifically, if the minimum bus voltage is reached it requires the crew to transition to a section in the procedure to remove the battery from service.

Part 2:

1) It can NOT be answered solely by knowing < 1 hour Tech Specs
2) It can NOT be answered solely by knowing the LCO/TRM information listed "above-the-line"
3) It can NOT be answered by knowing the Tech Spec Safety Limits or their bases
4) It requires the applicant to have knowledge contained in the Tech Spec Basis (specifically the minimum voltage limit for battery operability) to answer the question correctly.

2010 MNS SRO NRC Examination QUESTION 88

2588

B

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	MNS Exam Bank Question AP15N01

Development References
Learning Objectives: 1) AP15003 References: 1) Tech Spec 3.8.4 Basis (DC Sources - Operating) 2) AP-15 (Loss of Vital or Aux Control Power)

Student References Provided

KA	KA_desc
APE058	Ability to determine and interpret the following as they apply to the Loss of DC Power: (CFR: 43.5 / 45.13)□125V dc bus voltage, low/critical low, alarm .....
AA2.02	

401-9 Comments:

Remarks/Status

Question 88 References:

From Tech Spec 3.8.4 Basis:

BASES

BACKGROUND (continued)

Each battery (EVCA, EVCB, EVCC, EVCD) has adequate storage capacity to carry the required duty cycle for one hour after the loss of the battery charger output. In addition, the battery is capable of supplying power for the operation of anticipated momentary loads during the one hour period.

Each 125 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each channel is located in an area separated physically and electrically from the other channel to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

The batteries for the channels of DC are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 125% of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 150% of required capacity. The individual cell voltage limit is 2.13 V per cell. The minimum battery terminal voltage limit is greater than or equal to 125 V while on float charge as discussed in the UFSAR, Chapter 8 (Ref. 4). The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each channel of DC has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 8 hours while supplying normal steady state loads discussed in the UFSAR, Chapter 8 (Ref. 4).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 6), and in the UFSAR, Chapter 15 (Ref. 7), assume that Engineered Safety Feature (ESF) systems are OPERABLE.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

From AP-15 (Loss of Vital or Aux Control Power):

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE If any distribution center voltage goes down to 105 - 107 Volts, the associated battery has met its duty cycle requirement. Further depletion of the battery may result in battery damage. Opening the associated battery breaker when this voltage is reached will completely deenergize the associated distribution center.

__ 17. **IF AT ANY TIME dispatched operator notifies Control Room that distribution center voltage reaches limit in table below, THEN GO TO indicated step to remove battery from service:**

DISTRIBUTION CENTER	VOLTAGE LIMIT	STEP
DCA	107 Volts	Step 19
DCB	107 Volts	Step 20
EVDA	105 Volts	Step 21
EVDB	105 Volts	Step 22
EVDC	105 Volts	Step 23
EVDD	105 Volts	Step 24

__ 18. **Do not continue unless directed by Step 17.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21. **IF AT ANY TIME EVDA Distribution Center reaches 105 volts, THEN evaluate performing the following:**	
__ a. Dispatch operator to open Distribution Center EVDA Compartment 2A (Battery EVCA Switch).	
__ b. **IF AT ANY TIME** control power is needed to operate 'A' Train breakers, **THEN** contact station management to evaluate aligning battery to breaker control power circuits only.	
c. Notify Unit 2 to perform the following:	
__ 1) **WHEN** EVDA deenergized, **THEN REFER TO** AP/2/A/5500/15 (Loss of Vital or Aux Control Power) as time allows.	
__ 2) Trip Unit 2 reactor.	
__ 3) **GO TO** EP/2/A/5000/E-0 (Reactor Trip or Safety Injection).	
__ d. **WHEN** EVDA deenergized, **THEN RETURN TO** Step 1 in body of procedure as time allows.	
__ e. Trip Unit 1 reactor.	
__ f. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).	
22. **IF AT ANY TIME EVDB Distribution Center reaches 105 volts, THEN evaluate performing the following:**	
__ a. Dispatch operator to open Distribution Center EVDB Compartment 2A (Battery EVCB Switch).	
__ b. Notify Unit 2 to **GO TO** AP/2/A/5500/15 (Loss of Vital or Aux Control Power).	
__ c. **RETURN TO** Step 1 in the body of the procedure.	

Question 88 Parent Question:

Question 672 AP15N01 AP15N01

1 Pt

Unit one was operating at 100% power when a total loss of onsite and offsite power occurred. Given the following events and conditions:

- 1EVDA is supplying normal full power loads,
- No battery charger is available,
- Systems operate normally

Which one of the following statements correctly describes the minimum length of time that bus 1EVDA is designed to sustain loads and what action will protect the DC bus loads?

A. After 1 hour, the vital battery bus breaker will open automatically when bus voltage falls to 105 volts.

B. After 1 hour, the vital battery breaker must be manually opened when bus voltage falls to 105 volts.

C. After 4 hours, the vital battery breaker will open automatically when bus voltage falls to 107 volts.

D. After 4 hours, the vital battery breaker must be manually opened when bus voltage falls to 107 volts.

Answer 672

Answer: B

Distracter Analysis:

A. Incorrect: the vital battery breaker does not automatically open
Plausible: partially correct - the design time for sustaining loads is 1 hour

B. Correct: below this value the battery could be damaged or components will begin to fail.

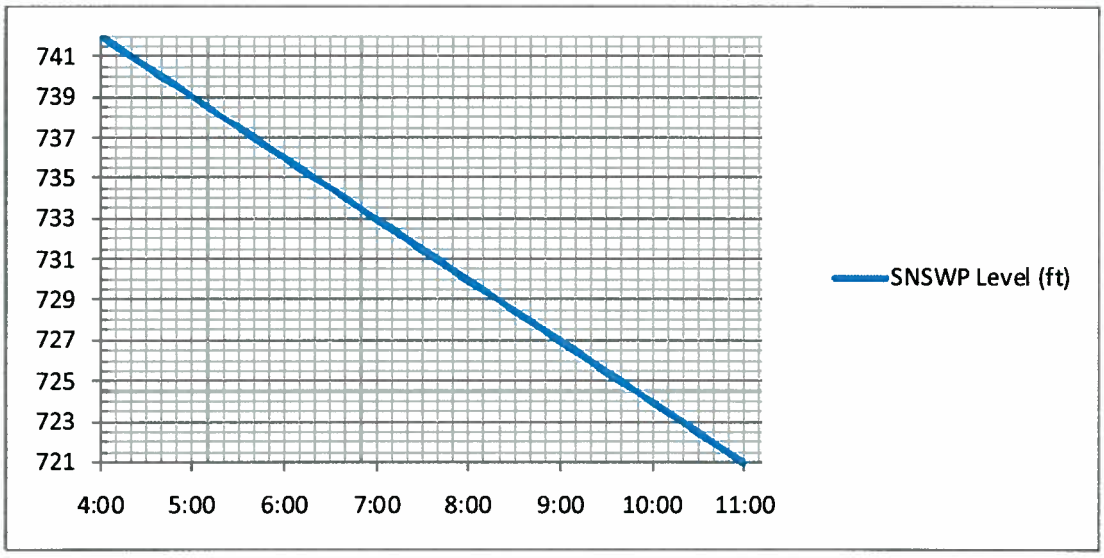
C. Incorrect: the battery is expected to last for 1 hour and there is no automatic trip associated with low voltage
Plausible: the 4 hour requirement for battery performance is typical of the aux batteries - voltage limit is 107 volts.

D. Incorrect: the vital batteries are not designed to sustain loads for 4 hours
Plausible: partially correct - DC bus protection is achieved by manually opening the breaker - voltage limit is 107 volts.

KA	KA_desc
APE062 2.4.47	APE062 GENERIC 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)

Unit 1 is operating at 100% RTP with "B" Train equipment in service when the following sequence of events occurs:

- The Low Level Intake suction has been lost due to fouling associated with the intake grating
- The crew is performing Enclosure 1 of AP-20 (Aligning B Train RN to Pond)
- 0RN-152 (Train 1B & 2B Disch to SNSWP) failed to open and all attempts to move the valve have failed
- The following SNSWP level trend is observed on the OAC:



Based on these conditions, the SNSWP level becomes INOPERABLE at __(1)__ .

The SNSWP minimum level ensures a sufficient volume of water to allow RN system operation for at least __(2)__ following a design basis LOCA.

Which ONE (1) of the following completes the statements above?

A. 1. 0450
   2. 5 Days

B. 1. 1040
   2. 5 Days

C. 1. 0450
   2. 30 Days

D. 1. 1040
   2. 30 Days

General Discussion

In the scenario the applicant is presented with a set of conditions where a loss LLI has resulted in the crew attempting to align the B Train of RN to the SNSWP. Due to the failure of a valve to move an abnormal alignment has resulted which is pumping water from the SNSWP to the lake and depleting the SNSWP inventory. Per TS 3.7.8 basis, the minimum required water level for the SNSWP is 739.5 which ensures a sufficient volume of water to allow NSWS to operate for 30 day following a design basis LOCA. With the SNSWP level trend provided in the stem, the applicant must differentiate between the TS level and the level at which SNSWP temperature is recorded (722').

Answer A Discussion

INCORRECT: See explanation above

PLAUSIBLE: Part 1 is correct and therefore plausible

Part 2 is plausible because 5 days is the time the D/G is designed to operate with the minimum fuel required available. The applicant may misinterpret this to imply that the safe shutdown loads supplied would be required to be available for the same time frame.

Answer B Discussion

INCORRECT: See explanation above

PLAUSIBLE: Part 1 is plausible if the applicant recalls that the 722 elevation is discussed in the Basis Document for TS 3.7.8 but confuses this elevation as that which is required for minimum level.

Part 2 is plausible because 5 days is the time the D/G is designed to operate with the minimum fuel required available. The applicant may misinterpret this to imply that the safe shutdown loads supplied would be required to be available for the same time frame.

Answer C Discussion

CORRECT: See explanation above.

Answer D Discussion

INCORRECT: See explanation above

PLAUSIBLE: Part 1 is plausible if the applicant recalls that the 722 elevation is discussed in the Basis Document for TS 3.7.8 but confuses this elevation as that which is required for minimum level.

Part 2 is plausible because 5 days is the time the D/G is designed to operate with the minimum fuel required available. The applicant may misinterpret this to imply that the safe shutdown loads supplied would be required to be available for the same time frame.

Basis for meeting the KA

KA is matched because the candidate must evaluate the provided indications and diagnose the time at which the indicated SNSWP level is below that which is assumed in safety analysis. The "utilizing the appropriate control room reference material" in this case would be the use of the OAC level trend for SNSWP level.

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. First the applicant must recall from memory the minimum level from the TS Basis assumed in the Safety Analysis. Then the applicant must analyze the SNSWP trend to determine at which time the SNSWP level decreases to less than the level recalled from memory.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs. There are no actions that are 1 hour or less associated with TS 3.7.8.
2) This question can NOT be answered by knowing information listed "above-the-line". The information solicited by the question is contained in the surveillance requirements for the spec and is therefore not "above-the-line" information.
3) This question can NOT be answered by knowing the TS Safety Limits or their bases. The information tested is from TS 3.7.8, SNSWP
4) This question requires the applicant to recall information contained in the TS basis for SNSWP 3.7.8. He must determine if the given indication meet the minimum required level assumed in safety analysis and also recall the mission time for the SNSWP to provide a water supply for the NSWS.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

2010 MNS SRO NRC Examination QUESTION 89

2589

C

Development References	Student References Provided
3.7.8 Basis AP-20 Enclosure 1	

KA	KA_desc
APE062	APE062 GENERIC 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)
2.4.47	

401-9 Comments:	Remarks/Status

Question 89 References:

From TS 3.7.8 Basis:

BASES

APPLICABLE SAFETY ANALYSES (continued)

cecay heat, and worst case single active failure (e.g., single failure of a manmade structure). The SNSWP is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the SNSWP.

The SNSWP satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

This is the basis containing the correct answer which is 30 days.

The SNSWP is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the NSWS to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the NSWS. To meet this condition, the SNSWP temperature should not exceed 82°F at 722 ft mean sea level and the level should not fall below 739.5 ft mean sea level during normal unit operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SNSWP is required to support OPERABILITY of the equipment serviced by the SNSWP be OPERABLE in these MODES.

In MODE 5 or 6, the requirements of the SNSWP are det systems it supports.

This section contains the minimum level which is 739.5 ft. The value for the leve for the required temperature is 722 ft which provide plausibility for the applicant selecting the distracter for adiquate.

ACTIONS

A.1

If the SNSWP is inoperable the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is

From AP-20 Enclosure 1

NOTE Shared RN valves can be operated from either unit's control switch.

1. **Align B Train RN to SNSWP as follows:**

__ a. Notify Unit 2 Operator that B Train RN will be aligned to the SNSWP.

__ b. Check 0RN-9B (Train 1B & 2B SNSWP Supply) - OPEN.

__ c. Open 0RN-152B (Train 1B & 2B Disch to SNSWP).

__ b. **GO TO** Case II (Loss of Low Level or RC Supply Crossover).

c. Perform the following to minimize pond depletion:

1) Close the following valves:

__ • 0RN-11B (Train 1B & 2B LLI Supply)

__ • 0RN-10AC (Train 1B & 2B LLI Supply).

__ 2) Dispatch operator to open 0RN-152B (Train 1B & 2B Disch to SNSWP) (aux bldg pipe chase, 716+2' EE-64, near Unit 2 containment sump lines 15' from cad door)

__ 3) **WHEN** 0RN-152B (Train 1B & 2B Disch to SNSWP) is open, **THEN** perform Steps 1.d through 1.h

__ 4) Monitor SNSWP level as required to prevent reducing SNSWP below Tech Spec level.

(RNO continued on next page)

KA	KA_desc
APE003	APE003 GENERIC□Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)
2.2.40	

Given the following conditions on Unit 1:

- Unit was at 100% RTP when rod M-4 dropped due to a blown fuse
- AP-14 (Rod Control Malfunction) has been implemented

1) In accordance with Tech Spec 3.1.4 (Rod Group Alignment Limits), if the rod can NOT be restored to within alignment limits, power must be reduced to less than or equal to _____ within 2 hours.

2) Per AP-14 power must be reduced to less than a MAXIMUM of _____ to retrieve the dropped rod.

Which ONE (1) of the following completes the statements above?

A. 1. 95% RTP
   2. 75% RTP

B. 1. 95% RTP
   2. 50% RTP

C. 1. 75% RTP
   2. 50% RTP

D. 1. 75% RTP
   2. 75% RTP

General Discussion

In accordance with Tech Spec 3.1.4 (Rod Group Alignment Limits) the misaligned rod (in this case a dropped rod) must be restored to alignment limits with 1 hour OR SDM must be verified with limits AND power reduced to less than 75% RTP within 2 hours.

In addition surviellances for Enthalpy Rise and Heat Flux Hot Channel factors must be performed within 72 hours.

In accordance with AP-14, power must be less than 50% RTP to retrieve the dropped rod. Additionally, AP-14 specifies power be reduced to less than 75% RTP within 2 hours to comply with Tech Spec 3.1.4.

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible because one of the actions required by Tech Spec 3.1.4 is to perform surveillance 3.2.2.1 for Enthalpy Rise Hot Channel factor determination. Normally this surveillance is required by Tech Spec 3.2.2 when power exceeds 95% RTP. Therefore it is plausible for the applicant to conclude that power needs to be reduced to less than this power so that the Enthalpy Rise Hot Channel factor surveillance can be performed.

Part 2 is plausible because power must be reduced to less than 75% RTP to comply with Tech Spec 3.1.4 if the rod can not be realigned within 2 hours.

Answer B Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible because one of the actions required by Tech Spec 3.1.4 is to perform surveillance 3.2.2.1 for Enthalpy Rise Hot Channel factor determination. Normally this surveillance is required by Tech Spec 3.2.2 when power exceeds 95% RTP. Therefore it is plausible for the applicant to conclude that power needs to be reduced to less than this power so that the Enthalpy Rise Hot Channel factor surveillance can be performed.

Part 2 is correct.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is correct.

Part 2 is plausible because power must be reduced to less than 75% RTP within 2 hours to comply with TS 3.1.4. It is plausible for the applicant to conclude that the same restriction for reducing power due to the misaligned rod applies to realigning the rod as well.

Basis for meeting the KA

Tech Spec 3.1.4 (Rod Group Alignment Limits) is the only TS that has applicability during a dropped rod scenario. Most of the actions contained in TS 3.1.4 are one hour or less actions making them RO level knowledge. One of the few actions that is not a 1 hour or less action requires the crew to "reduce power to less than or equal to 75% RTP within 2 hrs" if the misaligned rod (dropped rod in this case) can not be restored to within limits within 1 hour. The applicant demonstrates the ability to apply Tech Spec 3.1.4 (Rod Group Alignment Limits) by recalling from memory the specific actions required to comply with the spec.

Basis for Hi Cog

Basis for SRO only

Part 1 of this question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.
2) This question can NOT be answered by knowing information listed "above-the-line". This information is contained in the action statement section of TS 3.1.4.
3) This question can NOT be answered by knowing the TS Safety Limits or their bases. This question relates to TS 3.1.4 (Rod Group Alignment Limits)
4) This question requires the applicant to have knowledge of actions required in the application of Tech Spec 3.1.4 (specifically the power reduction required to comply with the spec)

Part 2 of this question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

2010 MNS SRO NRC Examination QUESTION 90

2590

C

1) The question can NOT be answered solely by knowing systems knowledge. The requirements for retreiving a dropped rod are not discussed in limits and precautions or in the systems lesson plan. Therefore, this is NOT systems knowledge.
2) The question can NOT be answered by knowing immediate operator actions. The action to reduce power to less than 50% RTP to recover the dropped rod is not an immediate action.
3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.
4) The question can NOT be answered solely by knowing the purpose, overal sequence of events, or overall mitigative strategy of the procedure. This is detailed knowledge of procedure content.
5) The question requires the applicant to have detailed procedure step knowledge from AP-14 (specifically the power reduction required to retrieve the dropped rod). Therefore, it is SRO knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References

Learning Objective:
1) AP-14 #4

References:
1) Lesson Plan OP-MC-IC-IRE

Student References Provided

KA	KA_desc
APE003	APE003 GENERIC Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)
2.2.40	

01-9 Comments:

Remarks/Status

Question 90 References:

From AP-14:

MNS AP/1/A/5500/14 UNIT 1	ROD CONTROL MALFUNCTION Enclosure 1 - Page 5 of 23 **Response To Dropped or Misaligned Rod**	PAGE NO. 9 of 37 Rev. 13

12. **Reduce reactor power below 50% prior to rod realignment as follows:**

__ a. Check only one rod - MISALIGNED.

__ a. GO TO Step 12.c.

__ b. Ensure reactor power is less than 75% within 2 hours of rod misalignment to comply with Tech Spec 3.1.4.

__ c. Reduce load as directed in subsequent steps until reactor power is less than 50% to comply with Reactor Engineering requirements.

d. Observe the following limitations during power reduction:

__ 1) Do not move rods until IAE determines rod movement is available.

__ 2) Borate as required during power reduction to maintain T-Avg at T-Ref.

__ 3) Monitor AFD during load reduction.

4) **IF AT ANY TIME** AFD reaches Tech Spec limit **AND** reactor power is greater than 50%, **THEN** perform the following:

__ a) Trip Reactor.

__ b) GO TO EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

e. Reduce reactor power to less than 50% **PER** one of the following procedures:

__ □ OP/1/A/6100/003 (Controlling Procedure For Unit Operation), Enclosure 4.2 (Power Reduction)

OR

__ □ AP/1/A/5500/04 (Rapid Downpower).

From TS 3.1.4 Basis:

ACTIONS (continued)

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour.

The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors

$F_Q(X,Y,Z)$ and $F^N_{\Delta H}(X,Y)$ must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 7). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(X,Y,Z)$ and $F^N_{\Delta H}(X,Y)$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(X,Y,Z)$ and $F^N_{\Delta H}(X,Y)$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of

From TS 3.1.4:

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits.	1 hour
	OR	
	B.2.1.1 Verify SDM is within the limit specified in the COLR.	1 hour
	OR	
	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	B.2.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND	
	B.2.3 Verify SDM is within the limit specified in the COLR.	Once per 12 hours
	AND	
	B.2.4 Perform SR 3.2.1.1.	72 hours
	AND	
	B.2.5 Perform SR 3.2.2.1.	72 hours
	AND	
	B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

(continued)

KA	KA_desc
APE069 AA2.02	Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: (CFR: 43.5 / 45.13)□Verification of automatic and manual means of restoring integrity .....

Given the following conditions on Unit 1:

- The unit is in MODE 5 following a refueling outage
- PT/1/A/4200/002 C (Containment Closure / Integrity) is in effect
- Both trains of ND are in service
- Both ND pumps trip and cannot be restarted
- AP-19 (Loss of ND or ND System Leakage) has been implemented

Which ONE (1) of the following describes actions required by AP-19 based on the conditions above?

A. Notify the WCC SRO to dispatch Operators to isolate any open penetrations ONLY.

B. Evacuate Containment AND notify the WCC SRO to dispatch Operators to isolate any open penetrations.

C. Notify the Containment Closure Coordinator to initiate Containment closure ONLY.

D. Evacuate Containment AND notify the Containment Closure Coordinator to initiate Containment closure.

General Discussion

In accordance with AP-19, if containment closure is in effect, AP-19 will direct the Operators to evacuate containment, initiate a Site Assembly, and notify the Containment Closure Coordinator to initiate Containment closure.

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: It is plausible for the applicant to conclude that since the unit is in a refueling outage the WCC SRO has control over all work on containment penetrations and has the resources to get all open penetrations isolated. Since evacuating containment has nothing to do with containment isolation per se, it is plausible to conclude that evacuating containment is not required.

Answer B Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This is plausible because evacuating Containment is required. It is also plausible for the applicant to conclude that since the unit is in a refueling outage the WCC SRO has control over all work on containment penetrations and has the resources to get all open penetrations isolated.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This is a correct action however not complete actions. Since evacuating containment has nothing to do with containment isolation per se, it is plausible to conclude that evacuating containment is not required.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The applicant is given a changing set of conditions which constitute a loss of containment integrity (because Containment integrity was not initially required and after conditions change it is required). The applicant is required to know how containment isolation is accomplished under this condition. Therefore, the KA is matched.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered by knowing systems knowledge. How containment integrity is controlled during shutdown periods is not covered by any systems lesson plan. Therefore, this is not systems level knowledge.
2) The question can NOT be answered by knowing immediate Operator actions. There are no immediate actions in AP-19.
3) The question can NOT be answered by knowing entry conditions for the AP. The actions for isolating containment in AP-19 are independent of the entry conditions.
4) The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of AP-19.
5) The question requires the applicant to have knowledge of detailed procedure content from AP-19 (specifically the steps requiring Containment evacuation and containment closure). Therefore, this is SRO level knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References

Learning Objective:
1) AP19002

References:
1) AP-19, Loss of ND or ND System Leakage

Student References Provided

2010 MNS SRO NRC Examination QUESTION 91

2591

D

KA	KA_desc
APE069 AA2.02	Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: (CFR: 43.5 / 45.13)□Verification of automatic and manual means of restoring integrity .....

401-9 Comments:	Remarks/Status

Question 91 References:

From AP-19:

MNS AP/1/A/5500/19 UNIT 1	LOSS OF ND OR ND SYSTEM LEAKAGE	PAGE NO. 8 of 217 Rev 22

5. **Evaluate isolating containment as follows:**

a. Check both ND pumps - OFF.

a. Perform the following:

__ 1) IF leak size greater than 10 GPM, THEN GO TO Step 5.b.

__ 2) IF leak caused trip 2 alarm on any Containment or Unit vent EMF, THEN GO TO Step 5.b.

__ 3) IF AT ANY TIME both ND pumps off, THEN perform Step 5.b through 5.f.

__ 4) IF AT ANY TIME leak size greater than 10 GPM OR leak caused trip 2 alarm on any Containment or Unit vent EMF, THEN perform Steps 5.b through 5.f.

__ 5) GO TO Step 6.

b. Announce the following on page:

__ 1) Description of event.

__ 2) "All personnel evacuate Unit 1 containment."

__ c. Actuate containment evacuation alarm.

__ d. REFER TO RP/0/A/5700/011 (Conducting a Site Assembly, Site Evacuation, or Containment Evacuation) while continuing with this procedure.

__ e. Check PT/1/A/4200/002 C (Containment Closure) - IN EFFECT.

__ e. GO TO Step 6.

__ f. Notify Containment Closure Coordinator to initiate containment closure.

KA	KA_desc
APE076	APE076 GENERIC□Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)
2.4.11	

Given the following conditions on Unit 1:

- The unit is at 100% RTP
- AP-18 (High Coolant Activity) has been entered due to 1EMF-18 (Reactor Coolant Filter 1A) in Trip 2 alarm

Isotopic analysis of the NC system indicates the presence of Cobalt and Manganese which indicates that a __(1)__ event has occurred and the required action in accordance with AP-18 to reduce the activity in the NC system is to __(2)__.

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

A. 1. failed fuel
   2. place the Cation Bed demineralizer in service

B. 1. failed fuel
   2. increase letdown flow

C. 1. crud burst
   2. place the Cation Bed demineralizer in service

D. 1. crud burst
   2. increase letdown flow

General Discussion

From the basis document for AP-18:

"Isotopes like Iodine and Cesium would indicate failed fuel, while isotopes like Cobalt and Manganese would indicate a crud burst."

For failed fuel events, one of the actions to reduce coolant activity is to place the Cation Bed Demineralizer in service. For a crud burst the appropriate action is to increase letdown flow.

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant is not familiar with the isotopes that would differentiate between a crud burst and failed fuel or the required actions from AP-18 which are specific to a crud burst or failed fuel event.

Answer B Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant is not familiar with the isotopes that would differentiate between a crud burst and failed fuel or the required actions from AP-18 which are specific to a crud burst or failed fuel event.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant is not familiar with the isotopes that would differentiate between a crud burst and failed fuel or the required actions from AP-18 which are specific to a crud burst or failed fuel event.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The KA is matched by Part 2 of the question in that the actions listed are from AP-18 and are different for failed fuel as opposed to a crud burst. art 1 of the question asks for information from the Background Document for AP-18.

Basis for Hi Cog

This is a higher cognitive level question because it requires multiple mental steps. The applicant must first recall from memory (from the basis document) that the presence of Cobalt means that a crud burst has occurred. The applicant must then recall from memory the appropriate actions for a crud burst to reduce activity levels.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered by knowing systems knowledge alone.
Knowledge of the different isotopes which indicate failed fuel or a crud burst and the methods for reducing radiation levels associated with those events is not expected knowledge for ROs or SROs at MNS. Therefore, it is not systems level knowledge.
2) The question can NOT be answered by knowing immediate Operator actions. There are no immediate actions associated with AP-18.
3) The question can NOT be answered by knowing AOP or EOP entry conditions. Knowing the entry conditions for AP-18 does not allow the applicant to answer this question.
4) The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.
5) The question requires the applicant to have detailed knowledge from the AP-18 Basis document (specifically that the presence of Cobalt and Manganese indicate a crud burst) and detailed procedure content knowledge (i.e. the requirement to increase letdown as opposed to placing the Cation Bed demineralizer in service). Therefore, this is SRO level knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

Learning Objectives:
1) AP18003

References:
1) AP-18
2) AP-18 Background Document

Student References Provided

	KA_desc
APE076	APE076 GENERIC Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)
2.4.11	

401-9 Comments:	Remarks/Status

Question 92 References:

From AP-18 Background Document for procedure Step 3:

STEP 3

PURPOSE:

To ensure the mixed bed demineralizer that's normally in service is not depleted and to determine if the cause of the high activity is from a crud burst or from failed fuel.

DISCUSSION:

Step 3.a checks the decontamination factor (DF) of the mixed bed demineralizer. DF is the ratio of the Influent concentration divided by the Effluent concentration. The higher the DF, the more effective the demin for removing impurities. A DF of 100 is typical of a fresh mixed bed, and a DF of 10 or less is typical of a mixed bed near depletion. If the DF were low, it would be appropriate for Chemistry to request swapping to the standby mixed bed.

Step 3.b request Chemistry to run an isotopic analysis to determine cause of high activity. Since they already have an influent sample in hand for determining DF, it can be used for this purpose. Isotopes like Iodine and Cesium would indicate failed fuel, while isotopes like Cobalt and Manganese would indicate a crud burst.

REFERENCES:

Primary Chemistry Lesson Plan OP-MC-CH-PC

STEP 4:

PURPOSE:

Reduce redeposition of crud throughout the plant.

DISCUSSION:

At the normal letdown flow rate of 75 gpm, it takes almost 21 hours to pass one entire volume of reactor coolant through the NV System. But a letdown flow of 120 gpm will circulate one entire volume of reactor coolant in approximately 12 hours (at 120 gpm letdown flow, 50% of the crud is removed every 12 hours).

REFERENCES:
Primary Chemistry Lesson Plan OP-MC-CH-PC

From AP-18:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

B. **Symptoms**

- **"1EMF-48 REACTOR COOLANT HI RAD" alarm**
- **"1EMF-18 REACTOR COOLANT FILTER 1A" alarm**
- **"1EMF-19 REACTOR COOLANT FILTER 1B" alarm**
- **Chemistry sample results indicate an unexpected increase in NC System activity.**

C. **Operator Actions**

__ 1. **Place one Outside Air Pressure Filter train in service PER Enclosure 1 (Pressurizing the Control Room).**	
__ 2. **Check 1NV-127A (L/D Hx Outlet 3-Way Temp Cntrl) - ALIGNED TO DEMIN.**	__ **Align valve to "DEMIN" position.**
3. **Determine cause of high activity as follows:**	
__ a. Request Chemistry to check decontamination factor of mixed bed demineralizer.	
__ b. Notify Chemistry to perform an NC System isotopic analysis to determine if high activity is from a crud burst or failed fuel.	
__ 4. **IF AT ANY TIME it is determined that high activity is from crud burst, THEN raise letdown flow to 120 GPM PER OP/1/A/6200/001 A (Chemical and Volume Control System Letdown), Enclosure 4.5 (Establishing Maximum Normal Letdown).**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **IF AT ANY TIME it is determined that high activity is from failed fuel, THEN perform the following:**	
__ a. Ensure mixed bed demineralizer in service.	
__ b. Notify Chemistry to consult with Reactor Group and RP to determine if the cation bed demineralizer should be placed in service.	
__ c. **IF AT ANY TIME** Chemistry requests cation bed demineralizer be placed in service, **THEN** place in service **PER** OP/1/A/6200/001D (Chemical and Volume Control System Demineralizers), Enclosure 4.3 (Removing/Returning the Cation Bed Demineralizer from/to Service).	
__ d. **REFER TO** RP/0/A/5700/000 (Classification of Emergency).	
__ e. Notify Reactor Group to discuss high activity in NC System with General Office Nuclear Engineering.	
__ 6. **Notify Radwaste to ensure VCT H_2 purge flow is established.**	
__ 7. **REFER TO Tech Spec 3.4.16 (RCS Specific Activity).**	
__ 8. **WHEN station management determines Control Room pressurization no longer required, THEN secure PER OP/0/A/6450/011 (Control Area Ventilation/Chilled Water System), Enclosure 4.4 (Control Room Atmosphere Pressurization During Abnormal Conditions).**	

END

KA	KA_desc
WE03	WE03 GENERIC Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)
2.4.46	

Given the following conditions on Unit 1:

- A Reactor Trip and Safety Injection have occurred due to a Small-Break LOCA inside Containment
- Containment pressure peaked at 2.5 PSIG
- ES-1.2 (Post LOCA Cooldown and Depressurization) has been implemented
- Both ND pumps are running
- NC system pressure is 250 PSIG and decreasing slowly

Which ONE (1) of the following describes the FIRST FWST level and Containment Sump conditions that require stopping both ND pumps prior to swapping to the containment sump?

A. FWST level 200 inches
Both "CONT SUMP LEVEL GREATER THAN 2.5 FT" alarms are DARK

B. FWST level 260 inches
Both "CONT SUMP LEVEL GREATER THAN 2.5 FT" alarms are DARK

C. FWST level 200 inches
Both "CONT SUMP LEVEL GREATER THAN 2.5 FT" alarms are LIT

D. FWST level 260 inches
Both "CONT SUMP LEVEL GREATER THAN 2.5 FT" alarms are LIT

General Discussion

In accordance with ES-1.2, if FWST level decrease to less than 250 inches AND both "CONT SUMP LEVEL GREATER THAN 2.5 FT" alarms are DARK, and NS pumps are OFF, the ND pumps must be stopped prior to reaching 180 inches to prevent vortexing following suction transfer to the sump.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

INCORRECT. See explanation above.

PLAUSIBLE: The second part is correct.

First part is plausible because the level is less than the level at which FWST makeup is required.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: First part is correct.

Second part is plausible if the applicant does not understand the significance of the alarm being lit or not lit. In other words, if the applicant does not understand that there has to be sufficient inventory in the Containment Sump prior to swapover to prevent vortexing of the ND pumps, the second part is plausible. Additionally, the Containment Sump level alarms being LIT under these plant conditions is normal.

Answer D Discussion

INCORRECT. See explanation above.

PLAUSIBLE: First part is plausible because the level is less than the level at which FWST makeup is required.

Second part is plausible if the applicant does not understand the significance of the alarm being lit or not lit. In other words, if the applicant does not understand that there has to be sufficient inventory in the Containment Sump prior to swapover to prevent vortexing of the ND pumps, the second part is plausible. Additionally, the Containment Sump level alarms being LIT under these plant conditions is normal.

Basis for meeting the KA

The applicant must understand the significance of the Containment Sump level alarms relative to plant conditions to know that the ND pumps must be stopped if FWST level decreases below a minumum level and sufficient inventory does not exist in the Containment Sump at the time of swapover to prevent vortexing of the ND pumps.

Basis for Hi Cog

This is a higher cognitive level question because it requires multiple mental steps. First the applicant must analyze the data given to understand that NS pumps are not running (i.e. Containment pressure peaked at 2.5 PSIG). The applicant must then recall from memory that less than 250 inches with both CONT SUMP LEVEL GREATER THAN 2.5 FT alarms DARK requires tripping both ND pumps.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge. Securing the ND Pumps if the potential for vortexing exists upon reaching the swapover point is not addressed in the limits and precautions or in the Systems Lesson Plan. Therefore, this is not systems level knowledge.
2) The question can NOT be answered by knowing immediate operator actions. There are no immediate actions associated with ES-1.2.
3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.
4) The question can NOT be answered solely by knowing the purpose, overal sequence of events, or overall mitigative strategy of the procedure.
5) This is detailed knowledge of a procedure diagnostic step that requires specific actions to be taken if conditions are not met. Therefore, this is SRO level knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

Learning Objectives:
1) EPE1004

Student References Provided

2010 MNS SRO NRC Examination QUESTION 93

2593

A

References:
E1 Background Document

KA	KA_desc
WE03	WE03 GENERIC□Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)
2.4.46	

401-9 Comments:	Remarks/Status

Question 93 References:

From ES-1.2 Background Document:

STEP 6 Check 1ETA and 1ETB - ENERGIZED BY OFFSITE POWER

PURPOSE: To ensure that the vital 4160V AC busses are energized.

BASIS: If offsite power is available, station equipment should be aligned to the offsite source. If either vital bus is **NOT** energized from its offsite source, AP/1/A/5500/07 (Loss of Electrical Power) should be referenced to ensure the automatic loading of equipment on the bus (e.g., charging pumps, MD CA pumps, KC pumps, etc.). This AP also provides actions to realign offsite power to a vital bus when the offsite source becomes available, addresses maintaining DC busses, and other issues associated with a loss of power.

STEP 7 Place all Pzr heaters in manual and off.

PURPOSE: To turn off all Pzr heaters prior to restoring Pzr level in order to minimize NC heat input.

BASIS: This action, consistent with normal cooldown procedures, prevents Pzr heat inputs from being automatically initiated. This added heat would tend to keep the NC pressurized.

NOTE: If all NC pumps are off, the upper head region may void during NC System depressurization. This will cause Pzr level to rise rapidly.

PURPOSE: To alert the operator of possible void formation in the NC during the NC depressurization.

BASIS: As the NC system is depressurized, steam may form in the hotter regions on the NC system. Pzr level will rise rapidly as water displaced from these voided regions replaces steam in the pressurizer. If voiding occurs, the Pzr may fill with water within a few minutes. This note informs the operator of this condition so that the NC system depressurization can be stopped quickly to avoid a water solid pressurizer.

STEP 8 Check if ND pumps should be stopped: (CONTINUOUS ACTION)

PURPOSE: To stop the ND pumps if NC pressure is above their shutoff head to prevent damage to the pumps.

BASIS: Upon S/I initiation all safeguard pumps are started regardless of the possibility of high NC pressure with respect to the low-head S/I pump shutoff head. On low-head systems where the pump recirculates at low flow there is concern with long term operation at low flow rates. Shutdown of the pump when the NC pressure meets the criteria outlined in this step allows for future pump operability. However, if NC pressure goes below 286 psig the pumps will have to be manually restarted since no automatic signal is available.

Additional criteria for stopping ND pumps were added to the step. For some low temperature mode 3 scenarios (described in PIP M-04-5515), the existing ERG step would leave ND pumps running with suction on FWST. For these small break LOCA

events, NS does not actuate. If FWST level reaches 250 inches and inadequate sump level is indicated, ND pumps must be secured prior to auto swapover to prevent them from vortexing. 250 inches was selected to provide many minutes for operators to respond. This step will also energize ND discharge valves and allow using them to isolate ND if single failure occurs preventing securing of ND pump. As documented in PIP M-04-5115, corrective action 11, ND operation for 10 minutes is always enough to ensure core reflood for an event initiated in Mode 3. By the time 250 inches FWST level is reached, ND operation much longer than this is assured. The TSC is requested to help monitor FWST level, since there is no alarm at 250 inches.

STEP 9 Control intact S/G levels: (CONTINUOUS ACTION)

PURPOSE: To ensure adequate feed flow or S/G inventory for secondary heat sink requirements.

BASIS: The minimum feed flow requirement satisfies the feed flow requirement of the Heat Sink Status tree until level in at least one S/G is restored into the narrow range. Narrow range level is reestablished in all S/Gs to maintain symmetric cooling of the NC. The control range ensures adequate inventory with level readings on span.

STEP 10 Initiate NC System cooldown to Cold shutdown:

PURPOSE: To begin or continue a controlled NC cooldown to cold shutdown using a preferred or alternate method with a specified maximum cooldown rate.

BASIS: The objective of a controlled cooldown is to reduce the overall temperature of the NC coolant and metal to reduce the need for supporting plant systems and equipment required for heat removal. The maximum cooldown rate of 100°F/hr will preclude violation of the Integrity Status Tree thermal shock limits. The preferred steam release path is to the condenser to conserve inventory; however, atmospheric release is the stated alternative. The ND system may have been placed in RHR mode later in the procedure, and should be used to cool down the NC to cold shutdown.

STEP 11 Check NC subcooling based on core exit T/Cs - GREATER THAN 0°F.

PURPOSE: To determine if the NC is subcooled so that subsequent actions dependent upon subcooling can be performed.

BASIS: If NC subcooling can be verified, the LOCA is most likely small and controllable, i.e., S/I flow equals or exceeds break flow. Subsequent steps that may be allowed include deliberate NC depressurization, NC pump restart, and S/I flow reduction. If subcooling is inadequate the operator is directed to increase S/I flow to restore subcooling.

KA	KA_desc
GEN2.1 2.1.4	Conduct of Operations□Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc. (CFR: 41.10 / 43.2)

Unit 1 is operating at 100% RTP.

An active licensed STA may assume the duties of the Control Room Supervisor provided the CRS or relief SRO is available to return to the control room within (1) AND the periods during which the STA assumes SRO duties do not exceed (2) in duration.

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

A. 1. 10 minutes
   2. 15 minutes

B. 1. 15 minutes
   2. 10 minutes

C. 1. 15 minutes
   2. 15 minutes

D. 1. 10 minutes
   2. 10 minutes

General Discussion

Technical Specifications allows the Shift Technical Advisor to assume the control room command function and perform the duties of the control room SRO in Modes 1, 2, 3, and 4 during periods when the CRSRO and the relief SRO are required to be absent from the control room. However, the following requirements must be met:

- The STA must hold an SRO license for the unit.
- The CRSRO or relief SRO must be available to return to the control room within 10 minutes.
- The periods during which the STA may perform the control room SRO duties may not exceed 15 minutes in duration or a total of 1 hour for the entire shift.

Answer A Discussion

CORRECT.

Answer B Discussion

Incorrect. Plausible if the candidate confuses the time for the CRSRO or relief SRO to return to the control room with the allowable duration of the relief by the STA.

Answer C Discussion

Incorrect. Plausible if the candidate confuses the time for the CRSRO or relief SRO to return to the control room with the allowable duration of the relief by the STA.

Answer D Discussion

Incorrect. Plausible if the candidate confuses the time for the CRSRO or relief SRO to return to the control room with the allowable duration of the relief by the STA.

Basis for meeting the KA

KA is matched because the candidate must understand the control room manning requirements for the individual fulfilling the control room command function.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(1 & 2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs. These requirements are in 5.1.2 which has no action statements.
2) This question can NOT be answered by knowing information listed "above-the-line". These are administrative requirements. There is no "above-the-line" knowledge.
3) This question can NOT be answered by knowing the TS Safety Limits or their bases. This is TS 5.1.2. not TS Safety Limits.
4) This question requires the applicant to have knowledge of TS administrative requirements contain in Section 5 of Tech Specs. This is SRO level knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	2009 MNS SRO Exam

Development References

Learning Objective:
1) OP-MC-ADM-OMP, Obj 3

References:
1) Technical Specification 5.1.2, amendment 213 and 194

Student References Provided

KA	KA_desc
GEN2.1	Conduct of Operations□Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc. (CFR: 41.10 / 43.2)
2.1.4	

401-9 Comments:

Remarks/Status

Question 94 References:

From T.S. 5.1.2:

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Station Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Control Room Senior Reactor Operator (CRSRO) shall be responsible for the control room command function. During any absence of the CRSRO from the control room while the unit is in MODE 1, 2, 3, or 4, an individual [other than the Shift Technical Advisor (STA)] with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the CRSRO from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

On occasion when there is a need for both the CRSRO and the relief SRO to be absent from the control room in MODE 1, 2, 3, or 4, the STA shall be allowed to assume the control room command function and serve as the SRO in the control room provided that:

a. the CRSRO or the relief SRO is available to return to the control room within 10 minutes.

b. the assumption of SRO duties by the STA is limited to periods not in excess of 15 minutes duration and a total time not to exceed 1 hour during any shift, and

c. the STA has a SRO license on the unit.

Question 94 Parent Question:

Examination Outline Cross-reference:	Level	RO	SRO
			X
	Tier #		3
Final	Group #		
	K/A #	G2.1.5	
	Importance Rating		3.9

Conduct of operations

Ability to locate and use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Proposed Question: SRO 86

1 Pt

Unit 1 is operating at 100% RTP.

Under which ONE (1) of the following conditions may an active licensed STA assume the duties of the Control Room Supervisor?

The CRS or relief SRO is available to return to the control room within ____(1) AND the periods during which the STA assumes SRO duties do not exceed __(2) in duration.

A. (1) 15 minutes
(2) 15 minutes

B. (1) 15 minutes
(2) 10 minutes

C. (1) 10 minutes
(2) 15 minutes

D. (1) 10 minutes
(2) 10 minutes

Proposed Answer: C

Explanation (Optional):

Technical Specifications allows the Shift Technical Advisor to assume the control room command function and perform the duties of the control room SRO in Modes 1, 2, 3, and 4 during periods when the CRSRO and the relief SRO are required to be absent from the control room. However, the following requirements must be met:

- The STA must hold an SRO license for the unit.
- The CRSRO or relief SRO must be available to return to the control room within 10 minutes.
- The periods during which the STA may perform the control room SRO duties may not exceed 15 minutes in duration or a total of 1 hour for the entire shift.

A. **Incorrect:** See explanation above. **Plausible** if the candidate confuses the time for the CRSRO or relief SRO to return to the control room with the allowable duration of the relief by the STA.
B. **Incorrect:** See explanation above. **Plausible** if the candidate confuses the time for the CRSRO or relief SRO to return to the control room with the allowable duration of the relief by the STA.
C. **Correct.**
D. **Incorrect:** See explanation above. **Plausible** if the candidate confuses the time for the CRSRO or relief SRO to return to the control room with the allowable duration of the relief by the STA.

Technical Reference(s) Technical Specification 5.1.2, amendment 213 and 194 (Attach if not previously provided)

(Including version or revision #)

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-MC-ADM-OMP, Obj 3 (As available)

Question Source:	Bank #	_____	
	Modified Bank #	_____	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	_____	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	_____	
10 CFR Part 55 Content:	55.41	_____	
	55.43	43.5	

Comments:

Conduct of operations

Ability to locate and use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

KA is matched because the candidate must understand the control room manning requirements for the individual fulfilling the control room command function.

This is an SRO Only question linked to 10CFR55.43(b)(2), Tech Specs. This questions can NOT be answered by knowing less than 1 hour Tech Spec or TRM action statements. It can NOT be answered by knowing the LCO/TRM information listed "above-the-line" (since this is an Administrative Control). It can NOT be answered by knowing Tech Spec Safety Limits or their basis. The candidate must apply requirements from Section 5.0, Administrative Controls of Technical specifications. Requirements in Section 5.0 are NOT expected knowledge for ROs.

KA	KA_desc
GEN2.1	Conduct of Operations□Ability to coordinate personnel activities outside the control room. (CFR: 41.10 / 45.5 / 45.12 / 45.13)
2.1.8	

Given the following conditions on Unit 1:

- The unit is in a refueling outage
- Fuel movement is in progress
- A leak has developed which has caused level to drop in the spent fuel pool
- The Spent Fuel Pool Level Low computer alarm has actuated

In accordance with AP-40 (Loss of Refueling Canal Level), which ONE (1) of the following describes the FIRST action directed by the CRS to mitigate the current conditions?

A. Place the weir gate in position and inflate the seals.

B. Begin makeup to the pool from the Boric Acid Tank.

C. Move the fuel transfer cart to the reactor side and close 1KF-122 (Fuel transfer tube block valve).

D. Move the fuel transfer cart to the spent fuel (pit) side and close 1KF-122 (fuel transfer tube block valve).

General Discussion

In accordance with AP-40 the first action which will be directed by the CRS is to move the fuel transfer cart to the spent fuel pit side and close 1KF-122.

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible because this would be the correct answer if 1KF-122 could not be closed. However, the first attempt is to close 1KF-122.

Answer B Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible because the Operators are directed to make upt o the spent fuel pool in AP-40. However, the first action is to attempt to isolate the spent fuel pool from the refueling canal the preserve the water that is in the spent fuel pool. Also, makeup to the spent fuel pool is not normally done from the BAT.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant does not recall which side of the transfer tube the fuel transfer cart has to be located to close the block valve (1KF-122).

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The KA is matched because the applicant must have knowledge of local operator actions outside of the control room to be able to coordinate those activities.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered by knowing systems knowledge alone. This is detailed procedure content knowledge from AP-40 and AP-41.
2) The question can NOT be answered by knowing immediate Operator actions. There are no immediate operator actions associated with AP-40 or AP-41.
3) The question can NOT be answered by knowing AOP or EOP entry conditions. Knowledge of AP-40 entry conditions will not enable the applicant to correctly answer this question.
4) The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of AP-40 or AP-41.
5) The question requires the applicant to assess plant conditions and then prescribing a procedure or section of a procedure to mitigate the consequences of the event. Specific to this event, initial entry would be into AP-41 (Loss of SFP Cooling or Level). However, since 1KF-122 is open the operator is directed out of AP-41 and into AP-40 (Loss of Refueling Canal Level) where they are directed to perform the appropriate actions.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	MNS Exam Bank Question FHFCN014

Development References

Learning Objective:
1)

References:
) Lesson Plan OP-MC-FH-FC Section 3.2.2
) AP-40

Student References Provided

2010 MNS SRO NRC Examination QUESTION 95

2595

D

KA	KA_desc
GEN2.1 2.1.8	Conduct of Operations□Ability to coordinate personnel activities outside the control room. (CFR: 41.10 / 45.5 / 45.12 / 45.13)

401-9 Comments:	Remarks/Status

Question 95 References:

From Lesson Plan OP-MC-FH-FC Section 3.2.2:

The Symptoms include:

- EMF36 UNIT VENT GAS HI RAD alarm
- EMF38 CONTAINMENT PART HI RAD alarm
- EMF39 CONTAINMENT GAS HI RAD alarm
- EMF40 CONTAINMENT IODINE HI RAD alarm
- EMF42 FUEL BLDG VENT HI RAD alarm
- EMF16 CONTAINMENT REFUELING BRIDGE alarm (2 - EMF3 on Unit 2)
- EMF17 SPENT FUEL BLDG REFUEL BRDG alarm (2 - EMF4 on Unit 2)
- Gas bubbles originating from the damaged assemblies
- Visible evidence of damage with the potential of radioactive releases

Operator Actions

CAUTION **Damage to the rubber Reactor Vessel Cavity Seal may occur if an assembly is dropped on or near it.**

Announce on page. If in containment, **evacuate** containment, assemble in contaminated change room and refer to RP/0/A/5700/11, Conducting a Site Assembly, Site Evacuation, or Containment Evacuation. **Isolate** containment: stop VP fans, ensure VP valves close, stop any VQ release, ensure equipment hatch closed, ensure one airlock door closed, dispatch Operator to move conveyor to Spent Fuel Pool Building, dispatch Operator to close KF-122. If high containment radiation exists, place Aux Carbon Filters in service per OP. Place Refueling Cavity in purification per OP.

If in Spent Fuel Building, **evacuate** Spent Fuel Pool area, assemble in contaminated change room. **Isolate** Spent Fuel Pool area: Check if VF EXH BYP DAMPER closed lite lit, and if not, place it's control switch to "CLOSE", and close the doors to the Spent Fuel Pool area. Ensure KF purification loop in service per OP.

Refer to RP/0/A/5700/00, Classification of Emergency.

3.2.2 AP/1/A/5500/40, LOSS OF REFUELING CANAL LEVEL

The purpose is to provide actions in the event of loss of water in the refueling canal.

The Symptoms include:

- "Spent Fuel Pool Level Low" computer alarm
- Decreasing level in refueling canal
- "Incore Inst Room Sump Hi Level" alarm
- EMF16 CONTAINMENT REFUELING BRIDGE alarm (2 - EMF3 on Unit 2)
- EMF17 SPENT FUEL BLDG REFUEL BRDG alarm (2 - EMF4 on Unit 2)

Operator Actions

NOTE Any available core location may be used when lowering a fuel assembly during emergency conditions.

If fuel movement is in progress: lower any assembly in the reactor building crane to fully down in the core, any assembly in the spent fuel crane to fully down, and any assembly in the upender to fully down. If they won't lower otherwise, manually release the brake and hand crank the hoist down. **NOTE: The sequence for lowering the hoist manually should be to put the emergency handwheel on the end of the hoist motor, hold it steady, while another person screws in the brake release (star shaped knob on a threaded stud) which when threaded in forces the brake disengaged. Care should be taken to remove the handwheel before electric operation of the hoist motor. The upender is similar. The bridge and trolly brake release is a lever, otherwise similar.** Dispatch Operator to locally move fuel transfer cart to the spent fuel (pit) side. Stop FWST Pump and close FW-13, and dispatch Operator to locally close KF-122. If KF-122 cannot be closed, then notify RP to begin surveys, consider installing the weir gate, and isolate the Spent Fuel Building (VF in filter mode and doors closed). Evacuate nonessential personnel from containment and Spent Fuel Building.

Try to identify and correct the cause of decreasing level. Verify seal integrity and air pressure to the Rx Vessel cavity seal and the Rx Vessel nozzle inspection port seals, and if not, reestablish VI to seals. Dispatch an Operator to locally ensure the Refueling Cavity Drains are closed. Check the S/G Nozzle Dams. Refer to AP/19, Loss of ND or ND System Leakage, while continuing with this procedure.

Makeup to the canal per OP/1/A/6200/13. **CAUTION:** Makeup to the SFP could dilute NC system boron concentration.

Monitor the Spent Fuel Pool level. If it gets to minus two feet, stop the KF Pump and turn off the lights. Initiated makeup per OP. If pool level low enough for radiation hazard, makeup from RN.

Ensure Containment Integrity with equipment hatch and airlock doors closed. If time permits, turn off canal underwater lights before they become uncovered. If necessary due to increasing radiation levels, consider using ND or NS to transfer water from the containment sump to the FWST for additional makeup capability.

Refer to RP/0/A/5700/00, Classification of Emergency.

From AP-40:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

B. Symptoms

- **"SPENT FUEL POOL LEVEL LOW" computer alarm**
- **Level in refueling cavity going down**
- **"INCORE INST ROOM SUMP HI LEVEL" alarm**
- **1EMF-16 "CONTAINMENT REFUELING BRDG" alarm**
- **1EMF-17 "SPENT FUEL BUILDING BRDG" alarm.**

C. Operator Actions

__ 1. **Announce occurrence on page.**

__ 2. **Check - FUEL MOVEMENT IN PROGRESS.**

Perform the following:

__ a. **IF** any radioactive component is being handled in the spent fuel pool or refueling cavity, **THEN** have fuel handling crew lower component to fully down.

__ b. **IF** cavity level is dropping more than one inch per minute, **AND** 1FW-27A (Unit 1 FWST to ND Pumps Isol) is open, **THEN** initiate makeup **PER** Enclosure 3 (Refueling Cavity Makeup Using ND Pump) while continuing in this AP.

__ c. **GO TO** Step 4.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE Any available core location may be used when lowering a fuel assembly during emergency conditions.

3. **Contact fuel handling SRO to have fuel handling crew perform the following:**

__ a. Lower any fuel assembly in the reactor building manipulator crane to fully down in the core.	__ a. Release brake and hand crank hoist down.
__ b. Lower any fuel assembly in the spent fuel manipulator crane to fully down.	__ b. Release brake and hand crank hoist down.
__ c. Lower any fuel assembly in either upender to fully down.	__ c. Release brake and hand crank upender down.
__ d. Move fuel transfer cart to the spent fuel (Pit) side.	__ d. Release brake and hand crank transfer cart to spent fuel (Pit) side.
__ e. Lower any radioactive component in the spent fuel pool or refueling cavity to fully down.	e. Perform the following: __ • Reinstall component. OR __ • Place component as far below the water as safely possible.

__ 4. **WHEN fuel transfer cart is in the spent fuel bldg, THEN dispatch 2 operators to CLOSE 1KF-122 (Unit 1 Fuel Transfer Tube Isol) (spent fuel bldg, 780, PP-51, top of fuel pool at south east corner).**

__ 5. **Notify Containment Closure Coordinator to initiate containment closure PER PT/1/A/4200/002 C (Containment Closure).**

Question 95 Parent Question:

FHFCN014
1 Pt(s) Given the following conditions:

- Unit 1 is in a refueling outage.
- Fuel movement is in progress.
- A leak has developed which has caused level to drop in the spent fuel pool.
- The Spent Fuel Pool Level Low computer alarm has actuated.
- Pool was initially at normal level and area radiation at 7 mrem/hr.
- After 20 minutes the pool level has decreased further and area radiation is 18 mrem/hr.

Which one (1) of the following describes the operator response to the current conditions?

A. Begin makeup to the pool from the Boric Acid Tank. to restore level.

B. Move the fuel transfer cart to the reactor side and close 1KF-122 (Fuel transfer tube block valve).

C. Move the fuel transfer cart to the spent fuel (pit) side and close 1KF-122 (fuel transfer tube block valve).

D. Place the weir gate in position and inflate the seals.

Answer 6

Answer: C

MISCINFO: SRO Only SOURCE: SRO92
REFERENCES: AP/1/A/5500/40, p. 2,3

LESSON: OP-MC-FH-FCB TASK:
p. 59, 60

OBJECTIVE: 15C TIME:
K/A: 000036G010 (3.7/3.8) DATE: SROEXAM1992

KA	KA_desc
GEN2.2	Equipment Control□Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)
2.2.40	

While the Unit is in MODE 1 Shutdown Margin is verified by __(1)__.

The basis for ensuring Shutdown Margin is maintained in MODE 1 is to protect against the effects of __(2)__.

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

A. 1. performing a SDM calculation
   2. an ejected rod

B. 1. checking control rods above insertion limits
   2. an ejected rod

C. 1. performing a SDM calculation
   2. a steam line break

D. 1. checking control rods above insertion limits
   2. a steam line break

General Discussion

Per TS 3.1.1 bases: During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits."

Per TS 3.1.6 bases: The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Answer A Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 Student may recall that a SDM verification is required in Mode 1 but misinterpret what method is required per TS Basis. The actual calculation would be performed in Mode 1 if there was an issue with Rod Insertion Limits or an inoperable control rod.

Part 2 is correct and therefore plausible.

Answer B Discussion

CORRECT: See explanation above.

Answer C Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 Student may recall that a SDM verification is required in Mode 1 but misinterpret what method is required per TS Basis. The actual calculation would be performed in Mode 1 if there was an issue with Rod Insertion Limits or an inoperable control rod.

Part 2 is plausible because a SLB is discussed extensively in the SDM basis (TS 3.1.1) but may misapply this consideration to apply to Mode 1 as well as Mode 2.

Answer D Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is correct and therefore plausible.

Part 2 is plausible because a SLB is discussed extensively in the SDM basis (TS 3.1.1) but may misapply this consideration to apply to Mode 1 as well as Mode 2.

Basis for meeting the KA

For this question, the applicant must have knowledge of surveillances related to meeting SDM. There is no TS for SDM in MODE 1. However, SDM met be met at all times, even in MODE 1. The applicant demonstrates the apply Technical Specifications by demonstrating an in-depth understanding of how we verify SDM is being maintained in MODE 1 (insertion limits) and in MODES 2 and below (by SDM calculation). Therefore, the KA is matched.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs. This question is related to knowledge of TS surveillance requirements (for SDM surviellance in both MODE 1 and MODE 2 and below) that are greater than 1 hour specifications.
2) This question can NOT be answered by knowing information listed "above-the-line". Surveillance requirements are below-the-line knowledge.
3) This question can NOT be answered by knowing the TS Safety Limits or their bases. TS 3.1.1 SDM and TS 3.1.6 Control Bank Insertion Limits.
4) This question requires the applicant to have knowledge of the application of TS Surviellance requirements and the basis for those surveillance requirements. Therefore, this is SRO level knowledge.

This Question is linked to 10CFR55.43(b)(2) for SRO only because it requires the applicant to recall information contained in the TS basis for SDM (TS 3.1.1), specifically what limiting accident the LCO is required to address and the mode of applicability for the specific accident. The applicant is further required to recall information contained in the basis for TS 3.1.6 (Control Bank Insertion Limits) associated with how the

verification of rod insertion limits is used to ensure the requirements of TS 3.1.1 are met in Mode 1.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	2009 AUDIT Q96 (Bank 1396)

Development References	Student References Provided
References: 1) TS3.1.1 and Basis 2) TS 3.1.6 and Basis	

KA	KA_desc
GEN2.2	Equipment Control□Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)
2.2.40	

401-9 Comments:	Remarks/Status

Question 96 References:

From Tech Spec 3.1.1 (SDM) Basis:

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or trip of all shutdown and control rods, assuming that the single rod cluster control assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Rod Control System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Rod Control System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on a reactor trip.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can eventually be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

One limiting accident for the SDM requirements is based on a main steam line break (MSLB) in Mode 2, as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a break of a main steam line upstream of the Main Steam Isolation Valve initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post-trip return-to-power may occur; however, no fuel damage occurs as a result of the post-trip return-to-power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

A potentially more limiting MSLB accident could occur for a steam line break outside containment when in Mode 3 with the low pressurizer pressure signal for safety injection actuation blocked. In this scenario, feedwater would not automatically isolate and the peak heat fluxes associated with the return-to-power may increase to values significantly

APPLICABLE SAFETY ANALYSES (continued)

greater than those in the accident analysis (Ref. 2). Therefore, when safety injection is blocked, administrative controls on boron concentration are required to prevent a return-to-power following a steam line break.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition; and

c. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power. SDM satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

Transients which are made less severe by the rapid insertion of control rod negative reactivity are also affected by the magnitude of the SDM limit. This is because the safety analyses assume a change in the rate of insertion of this negative reactivity when the SDM limit is reached. While the SDM is less than the limit value, the negative reactivity from the

APPLICABLE SAFETY ANALYSES (continued)

control rods is assumed to be inserted as quickly as the rod worth vs. time curves shown in Reference 5. When the SDM limit value is reached, the rate of negative reactivity insertion is decreased so that it is only fast enough to compensate for any positive reactivity insertion, e.g., from the cooling of the fuel and moderator (which normally have negative temperature coefficients). This methodology is conservative in that it does not take credit in the safety analyses, even temporarily, for a SDM greater than the limit value.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 4) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 5). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2 with $k_{eff} \geq 1.0$, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6.

From Tech Spec 3.1.6 (Rod Insertion Limits) Basis:

BACKGROUND (continued)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will ensure that fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function are no more than those predicted in and allowed by the safety analyses.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in excessive fuel cladding failures in the event of a LOCA, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

1. specified acceptable fuel design limits, or

2. Reactor Coolant System pressure boundary integrity; and

b. The core remains subcritical after accident transients (except for steam line break accident).

This statement provides the plausibility for the distracters containing a steam line break as the basis for ensuring SDM.

2010 MNS SRO NRC Examination QUESTION 97

2597

C

KA	KA_desc
GEN2.2	Equipment Control□Knowledge of the process for making changes to procedures. (CFR: 41.10 / 43.3 / 45.13)
2.2.6	

Given the following conditions on Unit 1:

- The unit is in MODE 5 preparing for a unit startup after refueling
- You are the Unit 1 Control Room Supervisor
- A Temporary Test procedure is being run on the 1B Boric Acid pump
- The OATC points out that several steps in the TT procedure should be concurrent verification steps to be consistent with similar steps in other test procedures

In accordance with NSD 703 (Administrative Instructions for Technical Procedures) the change to the Temporary Test Procedure should be processed as a __(1)__ change and a 10 CFR50.59 Evaluation in accordance with NSD 228 (Applicability Determination) __(2)__.

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

A. 1. minor
   2. is required

B. 1. major
   2. is required

C. 1. minor
   2. is NOT required

D. 1. major
   2. is NOT required

General Discussion

In accordance with NSD 703 (Administrative Instructions for Technical Procedures) Section 703.4 (Criteria For Procedure Revisions and Changes) Step 4.4.3.e. on of the examples of changes that fit the definition of a minor procedure change is "Add/delete inspection/verification signatures (e.g. QC Hold Point, Concurrent Verification). Therefore, this change is a minor procedure change.

In accordance with NSD 703 Section 7.2 (Review of Minor Procedure Changes) Step 5 "An Applicability Determination (NSD 228) is NOT required for a minor procedure change. A 10 CFR50.59 review is part of the Applicability Determination.

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is correct.

Part 2 is plausible if the applicant does not recall that an Applicability Determination (and 50.59 Review) is not required for minor procedure changes. It is plausible for an applicant to conclude that an Applicability Determination in accordance with NSD 228 is required regardless of whether the change is a minor or major revision. Part of an Applicability Determination is deciding whether a 50.59 review is required. If the applicant concludes that an Applicability Determination is required they might also mistakenly conclude that a 50.59 Evaluation is also required.

Answer B Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible if the applicant does not recall the requirements of NSD 703 regarding the difference between a major and minor procedure change.

Part 2 is plausible because an Applicability Determination in accordance with NSD 228 is always required for a major revision and part of the Applicability Determination is deciding whether a 50.59 Evaluation is required. It is plausible that the applicant may mistakenly conclude that this means a 50.59 Evaluation needs to be performed.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

INCORRECT. See explanation above.

PLAUSIBLE: Part 1 is plausible if the applicant does not recall the requirements of NSD 703 regarding the difference between a major and minor procedure change.

Part 2 is correct. While an applicability determination is required for a major change, a 50.59 review may or may not be required for the change. If a procedure is on the exclusion list for 50.59 reviews, it is not required for a major revision.

Basis for meeting the KA

The KA is matched because the applicant must have knowledge of the Fleet Procedure requirements regarding changes to technical procedures.

Basis for Hi Cog

Basis for SRO only

This question meets the following examples for an SRO only question as described in the Clarification Guidance for SRO-only Questions Rev 1 dated 03/11/2010 for questions linked to 10CFR55.43(b)(3) (Facility licensee procedures required to obtain authority for design and operating changes in the facility):

* 10 CFR 50.59 screening and evaluation process
* Processes for changing the plant or plant procedures

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References

Learning Objectives:
1) ADM-OP #2

References:
1) NSD 703 Administrative Instructions for Technical Procedures

Student References Provided

	KA_desc
GEN2.2	Equipment Control□Knowledge of the process for making changes to procedures. (CFR: 41.10 / 43.3 / 45.13)
2.2.6	

401-9 Comments:	Remarks/Status

Question 97 References:

From NSD 703:

2. If the procedure change or revision does not alter the results, requirements, or methods by which a procedure is performed, go to 703.5 for processing a minor procedure revision or 703.7 for processing a minor procedure change.

3. The list below provides some examples of changes that fit the definition of a minor procedure revision or minor procedure change:

Note: This list is not intended to be all inclusive.

a. Incorporating previously approved changes.

b. Correct editorial errors (e.g., misspelled words, grammatical errors, typographical errors).

c. During the certification process, if the electronic files have formatting differences from the approved version (e.g., line endings, page endings, word wraps).

d. Correct, delete, or add information (e.g., numbering or references to steps, pages, enclosures, or procedures, work location information, references to other documents, notes, cautions, warnings):

- Adding, deleting, or correcting references to documents that are no longer applicable (e.g., Site Directive deleted in favor of an NSD).
- Changing the assigned part sequence numbers but the referenced part does not change (e.g., Stock Code Numbers).
- Changing the part number where an acceptable substitute has been identified under the Acceptable Substitute Program.

e. Correct, delete, or add nontechnical or administrative actions.

f. Add/delete inspection/verification signatures (e.g., QC Hold Point, Concurrent Verification).

g. Modify the format of a step or section, but not change the results:

- Rephrasing a step, without changing its scope or results, to clarify.
- Rephrasing to avoid ambiguous wording (e.g., moving from general to specific).
- Changing units in a procedure data sheet (e.g., Scale face is marked in inches, but the procedure specified readings in percent. Rather than requiring the technician to perform the conversion each time the procedure is performed, change the procedure to reflect field configuration.)

h. Reflect changes in administrative work practices that are not commitment items:

- Dividing Test Equipment into Test Equipment and Other Equipment.
- Deleting "Verify to Control Copy" step (now on PPR).
- Deleting step for Supervisor to NA steps before beginning procedure (now covered in NSD 704).
- Deleting or adding step to "Get key to open cabinet" (when cabinet is no longer locked or is being locked).

i. Document changes initiated/substantiated by other processes evaluated in accordance with NSD 228 (Applicability Determination):

- Equipment number changes evaluated under the engineering change process.

b. Appendix M may be added to and reformatted as necessary to make it more applicable to a specific group. However, information shall not be removed from this appendix.

c. Document retention requirements do not apply to Appendix M.

d. The Reviewer's signature on the Procedure Change Process Record indicates that the procedure has been reviewed in accordance with Appendix M.

5. An Applicability Determination (NSD 228) is NOT required for a minor procedure change.

6. The Reviewer shall perform a detailed line by line review of all information changed as follows:

a. Ensure information contained within the procedure change is accurate and complete and that sufficient documentation is required by the procedure change to ensure the intent of the procedure is met.

b. Ensure WARNINGS or CAUTIONS are used appropriately to minimize risk to personnel or equipment.

c. Ensure the step(s) affected by the change can be accomplished in the sequence written.

d. Ensure the step(s) affected by the change meets the current Technical Specifications, UFSAR, SLCs, NSDs, Site Directives, etc.

e. Ensure the step(s) affected by the change provides for smooth interaction between site groups and efficient utilization of site resources.

7. A review of a minor procedure change does not require an extensive review of material not changed. Review the remaining parts of the procedure NOT changed to verify the following:

a. No missing procedure steps or pages.

b. No obvious formatting problems created by the revision:

- Inappropriate page breaks
- Cautions, Warnings or Notes not on same page with applicable step
- Incorrect step numbering

c. Change was appropriately incorporated into ALL affected parts of the procedure.

8. The Reviewer shall determine the need for cross-disciplinary or additional reviews based on the following:

a. Response of a system under direct control of another group will be altered.

b. Steps in a procedure may affect the use or operation of equipment under another group's control.

c. Another group will be required to provide personnel to assist in the performance of a procedure.

d. In cases where specific disciplines or training is needed other than that of the Reviewer to ensure a complete technical review of the change.

e. Procedures which affect or involve the Site Emergency Plan shall be reviewed by the Site Emergency Planning Section. Procedures which involve Environmental Emergency Response plans (part of the Site Emergency Plan) shall also be reviewed by the Site Environmental Management Section.

KA	KA_desc
GEN2.3 2.3.12	Radiation Control□Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)

SLC 16.11.20 (Gas Storage Tanks) limits the quantity of radioactivity in each Waste Gas Decay Tank (WGDT).

The basis for this limit assures the amount of radioactivity released would be substantially lower than the dose guideline values of _____.

Which One (1) of the following completes the statement above?

A. 10 CFR 20 during routine WGDT releases.

B. 10 CFR 100 during routine WGDT releases.

C. 10 CFR 20 in the event of a WG System leak or failure.

D. 10 CFR 100 in the event of a WG System leak or failure.

General Discussion

This SLC considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity in each pressurized gas storage tank in the WASTE GAS HOLDUP SYSTEM to assure that a release would be substantially below the dose guideline values of 10 CFR Part 100 for a postulated event.

Answer A Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Answer is plausible because the "normal" release limits are delineated in 10CFR20 which provides the limits for what can routinely be released to the environment. It would reasonable for the applicant to misinterpret the basis of the limits for the quantity of radioactivity allowed to be stored in a WGDT to be contained in this document.

Answer B Discussion

INCORRECT: See explanation above.

PLAUSIBLE: First part of the answer is correct, 10CFR100 is the basis for the dose guidelines. Second part of the answer is plausible because it would be reasonable for the applicant to associate the limits in SLC 16.11.20 to be associated with routine releases.

Answer C Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Answer is plausible because the "normal" release limits are delineated in 10CFR20 which provides the limits for what can routinely be released to the environment. It would reasonable for the applicant to misinterpret the basis of the limits for the quantity of radioactivity allowed to be stored in a WGDT to be contained in this document.
Second part of the answer is correct.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

K/A is matched because the knowledge contained in the basis for this SLC requires the applicant to recall information that is directly related to the radiological safety principle of the protection of the public by limiting the potential release of radioactive gases which could affect the public. Knowledge of the controlling document (10CFR100) and the limiting condition is a condition of a SRO license required to operate the station.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the "Clarification Guidance for SRO-only Questions (Rev 1 dated 03/11/2010)" under the Screening Criteria for question linked to 10CFR55.43(b)(2) (Tech Specs):

1) It can NOT be answered solely by knowing < 1 hour Tech Specs.
2) It can NOT be answered solely by knowing the LCO/TRM information listed "above-the-line".
3) It can NOT be answered by knowing the Tech Spec Safety Limits or their bases.
4) It DOES require the applicant to have detailed knowledge of Tech Spec basis information to determine the correct answer.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	MNS Bank WEWGN04

Development References
OP-MC-WE-WG Obj. 6 SLC 16.11.20

Student References Provided

KA	KA_desc
GEN2.3 2.3.12	Radiation Control□Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)

D

2010 MNS SRO NRC Examination QUESTION 98 2598

401-9 Comments:	Remarks/Status

Question 98 References:

From OP-MC-WE-WG Objectives

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the Waste Gas (WG) System. WEWG001	X	X	X	X	X
2	Describe the system flowpath during normal operation, shutdown operation and waste gas discharge. WEWG002	X	X	X	X	X
3	List four components that discharge waste gas into the WG Header. WEWG003	X	X	X	X	X
4	List two types of non-radioactive waste gas discharged into the WG Header. WEWG004	X	X	X	X	X
5	List the WG Discharge Flow Controller (WG-160) trips. WEWG005	X	X	X	X	X
6	Concerning the Selected Licensee Commitments (SLC) related to the WG System:					
	• Discuss any commitments and their applicability.			X	X	X
	• For any commitments that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any commitment is (are) not met and any action(s) required within one hour.			X	X	X
	• Discuss the basis for a given commitment.					
	* **SRO only** WEWG007				X	*

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.20 Gas Storage Tanks

COMMITMENT The quantity of radioactivity contained in each gas storage tank shall be limited ≤ 49,000 Curies noble gases (considered as Xe-133).

APPLICABILITY At all times.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Quantity of radioactive material in tank not within limit.	A.1 Suspend all additions of radioactive material to the tank.	Immediately
	AND	
	A.2 Reduce the tank contents to within limit.	48 hours

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.20.1 Verify the quantity of radioactive material contained in each gas storage tank is within limit when radioactive materials are being added to the tank.	24 hours

BASES

This SLC considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity in each pressurized gas storage tank in the WASTE GAS HOLDUP SYSTEM to assure that a release would be substantially below the dose guideline values of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

REFERENCES

None

Parent Question WEWGN04

1 Pt SLC 16.11.20 limits the quantity of radioactivity in each Waste Gas Decay Tank (WGDT). What is the basis for this limit?

A. Assures the amount of radioactivity released would be substantially lower than the dose guideline values of 10 CFR 20 during routine WGDT releases.

B. Assures the amount of radioactivity released would be substantially lower than the dose guideline values of 10 CFR 100 during routine WGDT releases.

C. Assures the amount of radioactivity released would be substantially lower than the dose guideline values of 10 CFR 20 in the event of a WG System leak or failure.

D. Assures the amount of radioactivity released would be substantially lower than the dose guideline values of 10 CFR 100 in the event of a WG System leak or failure.

Answer 114

D

SLC 16.11.20
SRO only

KA	KA_desc
GEN2.3	Radiation Control□Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)
2.3.14	

Given the following conditions:

- A General Emergency has been declared
- All Emergency Response facilities are activated
- A non-licensed operator must be dispatched from the Operations Support Center to an area with an identified radiation field of 110 REM/hr in order to isolate the pathway for a large release to the environment
- The operator will be in the area for approximately 15 minutes
- All listed operators have volunteered and are fully aware of the risks involved

Which one of the following is the preferred operator to perform the task in accordance with RP/0/A/5700/004 (General Emergency) Enclosure 4.4 (Request for Emergency Exposure)?

A. Operator A is 26 years old and once in the past received a single acute dose of 26 REM TEDE.

B. Operator B is 52 years old and once in the past received a single acute dose of 27 REM TEDE.

C. Operator C is 29 years old and has a chronic lifetime dose of 18 REM TEDE.

D. Operator D is 49 years old and has a chronic lifetime dose of 19 REM TEDE.

General Discussion

Per RP-003, only a volunteer who is fully aware of the risks involved should be selected. All factors being equal, select volunteers above the age of 45 and those who normally encounter little exposure.

For this particular case, Operator D, although younger than Operator B, has received the lowest cumulative dose and additionally has never received an acute dose (which Operator B has).

Answer A Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant concludes that younger person is more physically fit and thus more capable than an older person of receiving an acute dose without significant physical damage. An acute dose should be a once in a lifetime event (per Radiation Protection Policy Manual, Policy II-02, Planned Special Exposure Dose Limits for Occupationally Exposed Personnel) and this person is not suitable with others available.

Answer B Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant concludes that the oldest person should be selected. However, an acute dose should be a once in a lifetime event (per Radiation Protection Policy Manual, Policy II-02, Planned Special Exposure Dose Limits for Occupationally Exposed Personnel) and this person is not suitable with others available.

Answer C Discussion

INCORRECT. See explanation above.

PLAUSIBLE: This answer is plausible if the applicant concludes that younger person is more physically fit and thus more capable than an older person of receiving an acute dose without significant physical damage, this would be correct since this individual has the lowest cumulative dose.

Answer D Discussion

CORRECT. See explanation above.

The difference bewteen 18 and 19 R cumulative lifetime dose is negligible and the older candidate should be chosen.

Basis for meeting the KA

The KA is matched because the applicant must have knowledge of the radiation protection hazards during an emergency condition and the criteria for selection of the Operator who is most appropriate to receive an acute radiation exposure.

Basis for Hi Cog

This is a higher cognitive level question because it requires multiple mental steps. It requires the applicant to first recall from memory the actue exposure limits and requirements from RP-004 and then compare each answer to the recalled memory and to all of the other answers to select the most appropriate individual to receive the acute exposure.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in the "Clarification Guidance for SRO-only Questions (Rev 1 dated 03/11/2010)" under the Screening Criteria for question linked to 10CFR55.43(b)(5) (Assessment and Selection of Procedures):

1) The question can NOT be answered by knowing systems knowledge.
2) The question can NOT be answered by knowing immediate Operator actions.
3) The question can NOT be answered by knowing entry conditions for any procedure.
4) The question can NOT be answered by knowing the purpose, overall sequence of events, or overall mitigative strategy of any procedure.
5) The question requires the applicant to have knowledge of the content of RP-004 General Emergency. Since this procedure would only be used by an SRO, the content of the procedure can NOT be considered RO level knowledge.

Therefore, this is an SRO level question.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	2010 NRC Q98 (Bank 2598)

Development References

References:
1) RP/0/A/5700/004 General Emergency

Student References Provided

2010 MNS SRO NRC Examination QUESTION 99

2599

D

KA	KA_desc
GEN2.3 2.3.14	Radiation Control□Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)

401-9 Comments:	Remarks/Status

Question 99 References:

From RP-004, Site Area Emergency:

Enclosure 4.4

Request for Emergency Exposure (a)

RP/0/A/5700/004
Page 1 of 1

Activity	Total Effective Dose Equivalent (TEDE)	Lens of Eye	Other Organs (b)
All	5 rem	15 rem	50 rem
Protecting Valuable Property	10 rem	30 rem	100 rem
Life saving or Protection of Large Populations	25 rem	75 rem	250 rem
Life saving or Protection of Large Populations (c)	> 25 rem	> 75 rem	> 250 rem

(a) Excludes declared pregnant woman.

(b) Includes skin and body extremities.

(c) Only on a volunteer basis to persons fully aware of the risks involved. All factors being equal, select volunteers above the age of 45 and those who normally encounter little exposure.

RP Badge No.	Name	Age	Employer	Signature of Individual

My signature indicates my acknowledgement that I have been informed that I may be exposed to the levels of radiation indicated above. I have been fully briefed on the task to be accomplished and on the risks of this exposure.

I, _____ acknowledge this planned Emergency Exposure _____.
(RPM or designee, signature or note of verbal authorization) Date/Time

I, _____ approve this planned Emergency Exposure at _____.
(Emergency Coordinator or EOF Director, signature or note of verbal authorization) Date/Time

Subsequent Radiation Protection Action:

- Determine need for medical evaluation
- Initiate reporting requirements per 10CFR20
- Copy to Individual's Exposure History File

KA	KA_desc
GEN2.4 2.4.40	Emergency Procedures / Plan Knowledge of SRO responsibilities in emergency plan implementation. (CFR: 41.10 / 43.5 / 45.11)

Given the following plant conditions:

- An Unusual Event was declared on Unit 2.
- Initial Notification to the States, Counties and the NRC has been completed.
- The Emergency Coordinator has just made the decision to upgrade the classification to an Alert

The NRC is required to be notified immediately but no more than __(1)__ after change of classification.

After the initial notification of the change in classification is made to the State and Counties, follow up notifications are required to be made every __(2)__ until the emergency is terminated.

Which ONE (1) of the following completes the statements above?

A. 1. 1 hour
   2. hour

B. 1. 1 hour
   2. 4 hours

C. 1. 15 minutes
   2. hour

D. 1. 15 minutes
   2. 4 hours

General Discussion

In the scenario given, the applicant is presented with a situation where an NOUE was declared and all required initial notifications to the State, Counties, and the NRC has been completed. Subsequently, an escalation to an Alert occurs and the applicant is asked to evaluate the current notification requirements both to the NRC and the affect on the requirement for follow up notifications. Per our procedure, RP/29 the follow-up notification requirement will change from 4 hours (NOUE) to a new requirement of 1 hour for the new Alert classification. The NRC notification procedure, RP/10 requires that the NRC be notified immediately but not more than 1 hour after a change in classification.

Answer A Discussion

CORRECT: See explanation above

Answer B Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is correct and therefore plausible.

Part 2 is plausible because the follow notification requirement for a NOUE is 4 hours which was in effect prior to the upgrade in classification.

Answer C Discussion

INCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because this is correct for every offsite agency except for the NRC. The applicant may misapply the 15 minute requirement to the NRC.

Part 2 is correct and therefore plausible.

Answer D Discussion

IINCORRECT: See explanation above.

PLAUSIBLE: Part 1 is plausible because this is correct for every offsite agency except for the NRC. The applicant may misapply the 15 minute requirement to the NRC.

t 2 is plausible because the follow notification requirement for a NOUE is 4 hours which was in effect prior to the upgrade in classification.

Basis for meeting the KA

The KA is matched because the applicant must have knowledge of the SRO (OSM) responsibilities for implementing the Emergency Plan (i.e. notification requirements to offsite agencies after an esculation in emergency classification).

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. The applicant must first evaluate all of the information provided and then apply multiple rules to a change in a given situation. This requires the applicant relate understanding the rules pertaining to offsite notification and apply them to a dynamic situation.

Basis for SRO only

This question is not tied to 10CFR50.43 (b) but can be classified as an SRO Plant Specific Example. This question requires additional knowledge required for the higher license level and is unique to the SRO/OSM position. At MNS it is the responsability of the SRO to complete the notifications to offsite agencies and NRC notification to the NRC in the event that an emergency is declared. Per Lesson plan OP- MC-EP-EMP (Emergency Plan) the objectives, #12 (Complete the ENF) and #14 (Complete the NRC event notification worksheet) are SRO ONLY objectives. (LPSO). Both the understanding of the requirements and the actual completion of the required paperwork along with the transmittal are SRO ONLY tasks at MNS.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References

Learning Objective:
1) EP-EMP #12 and #14

29 (Notifications to Offsite Agencies From The C/R)
c. 4.2 Pg 1 of 8

RP/10 (NRC Immediate Notification Requirements)
Enc 4.1 Pg 1 of 14

Student References Provided

2010 MNS SRO NRC Examination QUESTION 100

2600

A

KA	KA_desc
GEN2.4 2.4.40	Emergency Procedures / Plan□Knowledge of SRO responsibilities in emergency plan implementation. (CFR: 41.10 / 43.5 / 45.11)

401-9 Comments:	Remarks/Status

Question 100 References:

OP-MC-EP-EMP Obj: 12 & 14

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
10	Given the EPIP's and the emergency situation, classify the event.				X	X
11	Given the EPIP's and the emergency situation, provide the appropriate Protective Action Recommendations (PAR's).				X	X
12	Given the EPIP's and the emergency situation, complete the Emergency Notification Form.				X	X
13	Describe the use of the Selective Signaling Phone System to notify the State and County.	X	X	X	X	X
14	Given the EPIP's and the emergency situation, complete the appropriate portions of the procedure for an NRC Event Notification Worksheet.				X	X
15	State the requirements for Initial and Follow-up Notifications including: • Time Requirements • Agencies to be contacted			X	X	X

NOTE: New initial messages for higher classification upgrades are addressed in Enclosure 4.1. {PIP-M-01-3711}

1. Make follow-up notifications according to the following table:

Follow-up Notifications

1. Follow-up notifications to the State(s) and Counties must be made according to the following schedule:

   -For a NOUE, every 4 hours until the emergency is terminated. For ALERT, SAE, or GE every hour until the emergency is terminated.

   OR

   -If there is any significant change to the situation (make notification as soon as possible).

   OR

   -As agreed upon with an Emergency Management official from each individual agency. Documentation shall be maintained for any agreed upon schedule change. The interval for ALERT, SAE, and GE shall not be greater than 2 hours to any agency.

2. If a follow-up is due and an upgrade to a higher classification is declared, there is no need to complete the follow-up ENF. In this case, the offsite agencies must be notified that the pending follow-up is being superseded by an upgrade to a higher classification and information will be provided.

3. Follow-up messages in the General Emergency classification that involve an upgrade in PARs must be communicated to the offsite agencies as soon as possible and within 15 minutes.

2. Complete an Emergency Notification Form by one of the following:

   ☐ Obtain a preprinted ENF.

   OR

   ☐ Obtain a blank ENF.

4.1.1	Events Requiring IMMEDIATE NOTIFICATIONS: REPORTABLE EVENTS Corresponding 10CFR Section in Brackets [ ]	REPORTING TIME REQUIREMENTS	
4.1.1.1 [50.72a(1)(i)]	The declaration of any of the Emergency Classes specified in the McGuire Emergency Plan	4.1.1.1	Immediately after notification to state(s) and local government (counties) and not later than one hour after the time the Emergency Class was declared. Immediately report any change from one Emergency Class to another or a termination of the Emergency Class. (Use Enclosure 4.2.) See follow up requirements in section 4.1.6 of this procedure.
	or		
[50.72c(1)(ii)]	any change from one Emergency Class to another		
	or		
[50.72c(1)(iii)]	a termination of the Emergency Class		
4.1.1.2 [20.1906]	Events involving receiving and opening packages containing quantities of radioactive material in excess of a Type A quantity as defined in section 71.4 and appendix A to part 71 of this chapter when;	4.1.1.2	NOTE: Reporting under 10CFR20.1906 should be made as follows: the licensee shall immediately notify the final delivery carrier and by telephone and telegram, mailgram, or facsimile and the NRC Operations Center at 9-301-816-5100.
[20.1906d(1)]	1) Removable radioactive surface contamination exceeds the limits of section 71.87(I) of 10CFR20;		
	or		
[20.1906d(2)]	2) External radiation levels exceed the limits of section 71.47 of this chapter.		
4.1.1.3 [20.2201a(i)]	Any lost, stolen, or missing licensed material in an aggregate quantity equal to or greater than 1,000 times the quantity specified in appendix C to part 20 under such circumstance that it appears to the licensee that an exposure could result to persons in unrestricted areas.	4.1.1.3	Immediately after its occurrence becomes known to the licensee.
	or		
[20.2201a(ii)]	3) Within 30 days after the occurrence of any lost, stolen, or missing licensed material becomes known to the licensee, all licensed material in a quantity greater than 10 times the quantity specified in appendix C to part 20 that is still missing at this time.		