CALVERT CLIFFS NUCLEAR POWER PLANT

2014 NRC

INITIAL LICENSED OPERATOR

RO WRITTEN EXAM

KEY

Revision 1 (Incorporates Ops Reviewer Comments)

- #1 - Made changes to distracter "B" to make it clearly incorrect. Changed correct answer wording and made it answer "B", and old distractor "B" is now "C"
- #3 - Made editorial change ... added space in distracter "B"
- #4 - Minor editorial correction to the stem of the question. Changed "stay" to "remain" in "A" & "C" distracters
- #6 - Clarified wording in the stem of the question
- #7 - Minor editorial correction to the stem of the question Changed CCW to Component Cooling
- #9 – Changed stem of question and made previously correct answer "B" incorrect, changed wording on distractors "C" & "D" with "D" being correct.
- #10 - Changed wording on distractor "A" and re-ordered the remaining distractors so correct answer is now "B"
- #12 - Got rid of unnecessary wording
- #13 - Added clarifying information to the stem of the question
- #14 - Minor editorial correction to all four answer choices
- #19 - Minor editorial correction to the stem of the question
- #22 – Minor editorial correction to the stem of the question
- #23 - Minor editorial correction to the stem of the question
- #24 - Minor editorial correction to distractors "A" & "D"
- #25 – Minor changes to distractor "A" to make more incorrect and minor change to "B" to make more correct.
- #26 – Minor editorial correction to the stem of the question
- #27 – Changed stem of question to indicate not BOC, changed all distractors to match stem.
- #28 – Changed stem of question to initial start of RCP, changed all distractors to match the stem
- #29 - Minor editorial correction to the stem of the question
- #31 – Changed correct answer "B" to be more straightforward
- #32 - Minor editorial correction to the stem of the question
- #33 - Minor editorial correction to the stem of the question
- #34 – Changed distractor "B" & "D" to SRW TB isolations vice Main HPSI Hdrs
- #36 - Minor editorial correction to the stem of the question
- #38 - Minor editorial correction to the stem of the question
- #41 - Minor editorial correction to the stem of the question
- #43 - Minor editorial correction to the stem of the question

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Revision 1 continued

- #46 - Minor editorial correction to the stem of the question
- #47 – Added ≥20 seconds to all answers
- #48 - Minor editorial correction to the stem of the question and answers "B & C"
- #51 - Minor editorial correction to the stem of the question and distractor "D"
- #52 – Changed "C" & "D" to say "Purge continues to operate"
- #53 – Changed correct answer to 1B D/G, reworded distractors "A,C & D"
- #55 – Changed distractor "B" to more plausible distractor
- #57 – Minor editorial correction to the stem and changed wording on correct answer
- #58 - Minor editorial correction to the stem of the question
- #62 – Changed order of distractors
- #66 - Minor editorial correction to the stem of the question
- #67 – Reworded the correct answer
- #68 – Changed correct answer from 4 hrs to 3 hrs
- #68 – Added "≥" to all answers
- #70 - Minor editorial correction to the stem of the question
- #73 - Minor editorial correction to the stem of the question
- Numerous changes to distractor order to equalize correct answers:
  18-A, 19-B, 19-C, 19-D

Revision 2 (Incorporates Collegial Review Comments)

- #1 – Changed stem from switchyard troubleshooting to RPS troubleshooting
- #17 – Changes stem voltage to be more within normal values
- #19 – Changed stem of question to be clearer, all distractors changed to be in line with new wording of stem. Correct answer changed from C to B
- #23 – Minor wording in stem to clarify which switches
- #25 – Changed distractor "A" (removed >15 min)
- #30 – Minor change to stem to clarify
- #35 – Changes distractor "A" & "D" from 0.5-5psig to 0.5-8psig
- #37 – Changed wording from increase/decrease to raise/lower
- #38 – Minor changes to stem to clarify. Fixed error in explanation
- #60 – Changes to distractors "A" & "C" (removed "or Containment Purge), also minor change in format of stem
- #64 – Changed distractor "B" to make it more incorrect
- #68 – Minor changes to distractor "D" to clarify temp was high
- #72 – Removed using provided reference
- #73 – Changed distractor "A" to clarify and make more incorrect
- 18-A, 20-B, 18-C, 19-D

Revision 3 (Incorporates Exam Validator Comments)

- #1 - Replaced question with a modified Bank question to match KA
- #9 - Minor changes to Stem to clarify
- #18 - Added "Following an RFO"
- #20 - Removed SDM
- #32 - Changed to Main Steam Leak Event and changed indications to PAMS instruments
- #35 – Changed distractors "A" & C" slightly to clarify
- #39 - Added " remain" to answers B & D
- #43 - Bolded "Technical Specifications"
- #52 - Inserted "Saturated". Changed distractor for B&D to placing monitor in Level Cal vice putting fuel in a safe location
- #60 - Bolded "Prevent Exceeding"
- #61 - Capitalized "Fans"
- #65 - Changed distractor "B" to >1 minute
- #69 - Changed Stem to Double Valve Isolation vice hazardous energy
- #71 - Reworded distractors to clarify

Revision 4 (Incorporates 2nd Round Exam Validator Comments)

- #9 - Minor change to wording in stem to clarify
- #12 - Added "Isolation" to L/D CV's in stem
- #31 - Minor editorial change to stem to clarify
- #61 - Fixed grammatical error in distractor C
- #62 - Minor wording change to stem to clarify what question asking
- #75 - Changed 2nd part of distractor "D" to match distractor "A"

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Revision 5 (Incorporates Fleet Reviewer & NRC Comments)

- #1 – Minor editorial change to stem for ease of reading (replace "basis" with "reason")
- #2 – Minor editorial change added "significant" to stem
- #3 - Minor editorial change to stem (Remove "Given" at beginning of question)
- #6 – Minor change to stem (added "for given conditions") to ensure all incorrect answers are wrong
- #8 – Added level change to all 4 answers to more closely match the K/A, added NOP/NOT to stem
- #12 – Added explanation of high cognitive level to answer explanation
- #14 – Changed "B" and "C" distractors to equalize correct answers for exam
- #20 – Wrote new question based on feedback from NRC about K/A mismatch and potential 2 right answers
- #23 – Minor editorial change to stem to read easier (moved "fire effects" to end)
- #24 – Minor editorial change to stem (grammatical errors fixed)
- #25 – Added explanation of high cognitive level to answer explanation
- #29 – Changed distractors "C" & "D" to make more plausible but still wrong
- #31 – Reworded distractors "A" & "C" for similar wording. So correct answer is not only one worded a certain way
- #32 – Removed 1st sentence of stem, not needed
- #39 – Added "for 5 days" to stem to not have boron equalization occuring
- #41 – Added S/G water level in stem (NRC suggestion)
- #42 – Minor editorial change to stem to clarify only one answer is correct
- #46 – Minor editorial change to stem "continue operation" vice "raise power"
- #52 – Editorial change, added "automatically" to distractors "A" & "B"
- #57 – Minor wording change to answer "C", "build up" vice "left in"
- #63 – Minor editorial changes to stem and distractor "A" to make distractor plausible grammatically
- #67 - Added explanation of high cognitive level to answer explanation
- #68 - Added explanation of high cognitive level to answer explanation
- #72 – Changed question to a new question based on similarity to question on Audit Exam (different question # but very similar concept and wording)
- #73 – Minor editorial change to stem "directed by AOP-2A"

18-A, 20-B, 18-C, 19-D

35 Low Cog, 40 High Cog

1. 007 Reactor Trip - Stabilization - Recovery

U-1 is at 85% power and is ramping to full load when a loss of the 500KV Black Bus occurs.

While performing the Core and RCS Heat Removal Safety Function in EOP-0, the CRO notices T_{COLD} continuing to lower.

Which ONE of the following is the correct operator action and the reason for the action?

A. Reduce AFW flow due to excess flow from automatic initiation of steam driven pump

B. Shut both MSIV's due to the MSR second stage MOV's failing to close on loss of power

C. Reduce AFW flow due to excess flow from the automatic start of 13 AFW pump

D. Shut both MSIV's due to MTCV's failing to fully close on loss of power

Answer: B

Answer Explanation:

A. **Incorrect** - A loss of the 500 KV Black Bus does not cause excessive AFW flow; (Steam Train).

B. **Correct** - A loss of the 500 KV Black Bus will cause the loss of MCC's 106 & 116 which will cause an excessive cooldown due to MSR 2nd stages not being isolated during the trip.

C. **Incorrect** - A loss of the 500 KV Black Bus does not cause excessive AFW flow; (Motor Train)

D. **Incorrect** - A loss of the 500 KV Black Bus does not cause an automatic start of 13 AFW pump resulting in excessive AFW Flow.

Question 1 (Q95013)			
Topic:	U-1 Turbine Trip Alternate Actions		
Tier/Group:	1/1		
K/A Info:	007 - Reactor Trip • EK3 - Knowledge of the reasons for the following as they apply to a reactor trip: • EK3.01 - Actions contained in EOP for reactor trip		
RO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	EOP-0, Post-Trip Immediate Actions EOP-0, Post-Trip Immediate Actions Technical Basis		
Comments:	None		

2. 008 - Pressurizer Vapor Space Accident

Given the following conditions:

- Unit-1 at 100% power
- Pressurizer pressure is 2250 PSIA
- PZR backup and proportional heater controls are in AUTO
- 1-HS-100 (PZR pressure control) in the "Y" position
- 1-HS-100-3 (PZR Htr cutoff) in the "X+Y" position
- A significant steam leak develops on the sensing line for 1-PT-100Y

Which **ONE** of the following describes the expected PROPORTIONAL HEATER response?
Assume no operator action

A. Proportional heaters will operate at approximately 1/3 higher power than before the failure

B. Proportional heaters will operate at approximately 1/3 lower power level than before the failure

C. Proportional heaters will have maximum power level and the red lights will be illuminated

D. Proportional heaters will have maximum power level and the green lights will be illuminated

Answer: C

Answer Explanation:

A. **Incorrect** - Proportional heaters will operate at approximately 1/3 higher power than before the failure. Power to the heaters will go to maximum due to a sensed lower pressure.

B. **Incorrect** - Proportional heaters will operate at approximately 1/3 lower power level than before the failure. Power to the heaters will go to maximum due to a sensed lower pressure.

C. **Correct** - Proportional heaters will have maximum power and the red lights will be illuminated. Heaters will respond as if RCS pressure were low.

D. **Incorrect** - Proportional heaters will have maximum power and the red lights will be illuminated. Green/red lights are function of breaker positions, supply breakers remain shut.

Question 2 (Q20438)			
Topic:	Predict the response of Pressurizer controls to a PT-100Y failure		
Tier/Group:	1/1		
K/A Info:	008 - Pressurizer Vapor Space Accident • AK2 - Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: • AK2.03 - Controllers and positioners		
RO Importance:	2.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	Used in the 7/2002 NRC exam		
Technical references:	1C06-ALM, RCS Control Alarm Manual SD-064D, Reactor Coolant Instrumentation System Description		
Comments:	None		

3. 009 - Small Break LOCA

U-2 was operating at 100% power in a normal lineup prior to a Loss of Coolant Accident. Given the following:

- The reactor tripped 10 minutes ago
- RCS pressure is stable at 1270 PSIA
- PZR level is stable at 20 inches
- T_{COLD} is stable at 531°F
- 24 4KV bus is faulted
- No Operator actions have been taken

Which **ONE** of the following represents indicated net charging flow (charging flow - letdown flow)?

A. 44 GPM

B. 58 GPM

C. 88 GPM

D. 132 GPM

Answer: C

Answer Explanation:

A. **Incorrect** - 21 & 23 Charging Pumps are normally aligned to 21 bus and would both be running resulting a Charging flow of 88 GPM. This flow represents a single Charging Pp in operation which would be the case if 23 Charging Pump were aligned to 24 Bus.)

B. **Incorrect** - 21 & 23 Charging Pumps aligned to 21 bus and will be running (this flow is for 2 Charging Pumps - min letdown but letdown is isolated by SIAS)

C. **Correct** - 21 & 23 Charging Pumps are normally aligned to 21 bus and would both be running resulting in a charging flow of 88 GPM, letdown isolated by SIAS.

D. **Incorrect** - 21 & 23 Charging Pumps aligned to 21 bus and will be running (this flow is for 3 Charging Pumps)

Question 3 (Q97133)			
Topic:	Charging Flow during a Small Break LOCA		
Tier/Group:	1/1		
K/A Info:	009 - Small Break LOCA • EA2 - Ability to determine or interpret the following as they apply to a Small Break LOCA: • EA2.13 - Charging pump flow indication		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(8)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EOP-5, Loss of Coolant Accident SD-041, Chemical & Volume Control System AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power		
Comments:	None		

4. 011 - Large Break LOCA

What is the purpose of the LOCI Sequencer and which ESFAS signals are required for actuation?

A. Ensure D/G voltage and frequency remain in band during loading;
SIAS

B. Ensure ESF loads are all started to mitigate accidents:
SIAS

C. Ensure D/G voltage and frequency remain in band during loading;
SIAS & UV

D. Ensure ESF loads are all started to mitigate accidents;
SIAS & UV

Answer: C

Answer Explanation:

A. **Incorrect** - SIAS **and** UV Actuations are required to trigger operation of the LOCI Sequencer

B. **Incorrect** - Without the LOCI Sequencer all of the ESF loads would start simultaneously. The purpose of the LOCI Sequencer is to gradually load the DG to maintain voltage and frequency in band. SIAS **and** UV Actuations are required to trigger operation of the LOCI Sequencer

C. **Correct** - Without the LOCI Sequencer all of the ESF loads would start simultaneously. The purpose of the LOCI Sequencer is to gradually load the DG to maintain voltage and frequency in band. SIAS and UV Actuations are required to trigger operation of the LOCI Sequencer

D. **Incorrect** - Without the LOCI Sequencer all of the ESF loads would start simultaneously. The purpose of the LOCI Sequencer is to gradually load the DG to maintain voltage and frequency in band.

Question 4 (Q97159)			
Topic:	D/G basis for LOCA		
Tier/Group:	1/1		
K/A Info:	011 - Large Break LOCA • EK3 - Knowledge of the reasons for the following responses as they apply to the Large Break LOCA: • EK3.09 – Maintaining D/G's available to provide standby power		
RO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	SD-048, Engineered Safety Features Actuation System Description		
Comments:	None		

5. 015/017 - RCP Malfunctions

Given the following indications 10 minutes after a reactor trip with no operator actions:

- 11 S/G level -35" and 875 PSIA
- 12 S/G level -20" and 870 PSIA

Which **ONE** of the conditions below (assume no other malfunctions) would cause these indications?

A. 12A RCP locked rotor

B. Steam leak from 12 S/G

C. Steam leak from 11 S/G

D. 11 S/G tube leak of 120 GPM

Answer: A

Answer Explanation:

A. **Correct** - A locked rotor on 12A RCP causes reduced flow in 12 loop resulting in less heat transfer to 12 S/G. Therefore a slightly lower S/G pressure and a slightly higher S/G level exist in 12 S/G.

B. **Incorrect** - A steam leak from 12 S/G would cause a lower level in 12 S/G. Since 12 S/G level is higher than 11 S/G level this is incorrect.

C. **Incorrect** - A steam leak from 11 S/G would cause a lower pressure in 11 S/G. Since 11 S/G pressure is higher than 12 S/G pressure this is incorrect.

D. **Incorrect** - A tube leak on 11 S/G would cause a higher S/G level due to flow from the RCS into the S/G.

Question 5 (Q97160)			
Topic:	Unbalanced RCS flow thermodynamic relationship		
Tier/Group:	1/1		
K/A Info:	015/017 - RCP Malfunctions • AK1 - Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): • AK1.04 - Basic steady state thermodynamic relationship between RCS loops and S/Gs resulting from unbalanced RCS flow		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:			
Comments:	None		

6. 025 - Loss of RHR System

Given the following:

- Unit-1 is in Mode 6
- Refueling Pool level is 63 Ft. in preparation for UGS removal
- A control circuit malfunction causes 1-SI-652-MOV (SDC HDR RETURN ISOL) to shut
- ALL attempts to reopen 1-SI-652-MOV are unsuccessful

Which **ONE** of the following methods is directed, by procedure, to restore decay heat removal for these given conditions?

A. Use a SFP Cooling Pump to cool the Refueling Pool.

B. Initiate Once-Thru-Cooling using a HPSI Pump or LPSI Pump.

C. Open the SFP Transfer Gate Valve and cool with the SFP cooling system.

D. Fill the Refueling Pool using a HPSI Pump and makeup as inventory is lost to boil-off.

Answer: A

Answer Explanation:

A. **Correct** - Per AOP-3B, Section VII, for common mode loss of SDC with the RFP that is or can be filled.

B. **Incorrect** - These actions are AOP-3B directions when the RFP is **NOT** available.

C. **Incorrect** - The SFP Transfer Gate is verified shut in AOP-3B. The RFP level is lower than the SFP level such that opening the transfer gate would cause a loss of inventory in the SFP.

D. **Incorrect** - Boil-off of the Refueling Pool inventory is an option, but is not directed by AOP-3B for the given conditions.

Question 6 (Q92092)			
Topic:	Complete loss of SDC occurs w/RFP level at 63 Ft		
Tier/Group:	1/1		
K/A Info:	025 - Loss of RHR System • 2.1 - Conduct of Operations • 2.1.20 - Ability to interpret and execute procedure steps.		
RO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-3B, Abnormal Shutdown Cooling Conditions AOP-3B, Abnormal Shutdown Cooling Conditions Technical Basis		
Comments:	None		

7. 026 - Loss of Component Cooling Water

Unit-1 is operating at 100% power. 11 Component Cooling pump is in operation when a Component Cooling leak develops. The following conditions exist:

- Component Cooling Head tank level is 5" and slowly lowering
- Component Cooling Standby header pressure is 75 PSIG and slowly lowering
- 1C10 alarm "CNTMT NORMAL SUMP LVL HI" is in
- 1C07B alarm "12A RCP CCW FLOW LO" is in
- 12A Thrust Bearing and CBO temperatures are rising

Which **ONE** of the following actions is required based on existing plant conditions?

A. Shut the Component Cooling CVs to Containment, trip the reactor, and secure the RCPs.

B. Secure the operating Component Cooling pump due to cavitation caused by low head tank level

C. Restore Component Cooling Head tank level by opening the bypass to the makeup CV

D. Trip the Reactor since no Component Cooling pumps are operating.

Answer: A

Answer Explanation:

A. **Correct** - Based on the RCP information given, it can be deduced that a CCW leak exists on the 12A RCP. Isolating the leakage by shutting the containment isolation valves requires tripping the reactor then the RCPs once reactivity control safety function is verified.

B. **Incorrect** - Based on the standby header pressure, there is adequate pump pressure. Standby and normal CC header pressures are the same when the CCW system is in a normal alignment and does not depend on 11 or 12 CCW pump in operation. With a pressure of 75 PSIG, there is a CCW pump operating.

C. **Incorrect** - A CCW System leak exists, is given. The priority would be placed on isolating the leak. Once the leak is isolated, bypassing the make-up CV would only occur if the normal make-up source was not restoring CCW Head Tank level.

   Since RCP trip criteria will be met on after a loss of CCW flow, a manual trip would be required once the containment isolation CVs are shut.

D. **Incorrect** - Based on the standby header pressure, there is adequate pump pressure. Standby and normal CCW header pressures are the same when the CCW system is in a normal alignment and does not depend on 11 or 12 CCW pump in operation. With a pressure of 75 PSIG, there is a CCW pump operating.

Based on the simulator, the CCW pumps will not show signs of cavitation when head tank level is visible. This assumes that the pump is not in a runout condition, and a standby pressure of ~75 PSIG does not indicate runout conditions.

Question 7 (Q39973)			
Topic:	Response to a CCW leak in Mode 1		
Tier/Group:	1/1		
K/A Info:	026 - Loss of Component Cooling Water • AA1 - Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: • AA1.07 - Flow rates to the components and systems that are serviced by the CCWS; interactions among the components.		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-7C, Component Cooling System OM-51, U-1 Component Cooling System		
Comments:	None		

8. 027 - Pressurizer Pressure Control System Malfunction

While operating at NOP/NOT the Pressure transmitter associated with the selected Pressurizer Pressure Control Channel fails to 2210 PSIA.

Which **ONE** of the following statements represents the response of Pressurizer water temperature/level and why?

A. Temperature increase / level rises;
Proportional heaters provide maximum current

B. Temperature increase / level rises;
All Pressurizer heaters energized and providing maximum current

C. Temperature decrease / level lowers;
Pressurizer spray valves open causing saturation temperature to lower

D. Temperature decrease / level lowers;
All Pressurizer heaters providing minimum current with ambient losses causing temperature to lower

Answer: A

Answer Explanation:

A. **Correct** - Sensed pressure is below setpoint therefore proportional heaters fire fully causing temperature in PZR to rise

B. **Incorrect** - Backup heaters come on at 2200 PSIA, since sensed pressure is above 2200 PSIA the B/U heaters stay off

C. **Incorrect** - Sensed pressure is below setpoint therefore no additional spray (plausible if misinterpret controller response)

D. **Incorrect** - Sensed pressure is below setpoint therefore heaters energize not secure (plausible if misinterpret controller response)

Question 8 (Q97192)			
Topic:	Pressurizer response to a rise in RCS temperature		
Tier/Group:	1/1		
K/A Info:	027 - Pressurizer Pressure Control System Malfunction • AK1 - Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: • AK1.02 - Expansion of liquids as temperature increases		
RO Importance:	2.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(8)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:			
Comments:	None		

9. 029 - ATWS

An Automatic RPS trip signal failed to deenergize trip path relays K1 and K2. What action (if any) is required in accordance with the appropriate procedure to trip the reactor?

A. None, trip paths K3 and K4 will de-energize to trip the reactor.

B. A Manual trip is required by de-energizing the CEDM MG sets.

C. A Manual trip is required by depressing the pushbuttons on both 1C05 and 1C15

D. A Manual trip is required by depressing the pushbuttons from either 1C05 or 1C15

Answer: D

Answer Explanation:

A. **Incorrect** - Trip Path relays K1 and K2 maintain power to the CEAs, the reactor will not trip.

B. **Incorrect** - Trip Path relays K1 and K2 maintain power to the CEAs, the reactor will not trip. IAW EOP-0 pushbuttons on 1C05 or 1C15 will be depressed and the reactor will trip so de-energizing CEDM MG sets is not required.

C. **Incorrect** – Only the pushbuttons at 1C05 or 1C15 are required to be depressed to trip the reactor.

D. **Correct** - Only the pushbuttons at 1C05 or 1C15 are required to be depressed to trip the reactor.

Question 9 (Q20175)			
Topic:	Effect of trip paths K1 and K2 failing to deenergize on a reactor trip signal.		
Tier/Group:	1/1		
K/A Info:	029 - ATWS • EK2 - Knowledge of the interrelations between the and the following an ATWS: • EK2.06 - Breakers, relays, and disconnects		
RO Importance:	2.9*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	FSAR figure 7-2, RPS Functional Diagram		
Comments:	None		

10. 038 - Steam Generator Tube Rupture

When a Steam Generator tube rupture is suspected, Chemistry is called to perform qualitative samples on both SGs for activity **PER** CP-436 to confirm the affected S/G.

Which **ONE** of the following actions is performed first?

A. Check dose rates on the S/G blowdown piping

B. Check dose rates locally on main steam line piping

C. Draw a Condenser off-gas sample and analyze for activity

D. Draw grab samples from both S/G's and analyze for activity

Answer: B

Answer Explanation:

A. **Incorrect** - This action is not required by CP-436. Blowdown could be secured prior to Chemistry getting readings.

B. **Correct** - This action is the first specified in CP-436 and can distinguish which S/G contains activity

C. **Incorrect** - This is not specified in CP-436. It would supply adequate information to confirm a tube leak but could not be used to identify which Steam Generator had the leaking tube(s)

D. **Incorrect** - This is one of last actions in CP-436, only performed if no results gained from rad readings

Question 10 (Q97134)			
Topic:	SGTR diagnosis from local radiation readings		
Tier/Group:	1/1		
K/A Info:	038 - Steam Generator Tube Rupture • EA2 - Ability to determine or interpret the following as they apply to a SGTR: • EA2.11 - Local radiation readings on main steam lines		
RO Importance:	3.7*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EOP-6, Steam Generator Tube Rupture; CP-436, Qualitative Determination of Affected S/G in a Tube Leak Event		
Comments:	None		

11. 054 - Loss of Main Feedwater

During a loss of all feedwater event, EOP-3 directs action when _____ Steam Generator level(s) reach(es) <-350 inches to _______________.

A. Both, ensure OTCC in service prior to S/G dryout

B. One, ensure OTCC in service prior to S/G dryout

C. Both, protect the tubesheet and prevent a primary to secondary leak

D. One, protect the tubesheet and prevent a primary to secondary leak

Answer: A

Answer Explanation:

A. **Correct** - EOP-3 directs OTCC initiation when **BOTH** S/G's <-350 inches. EOP-3 Basis states that initiating OTCC **prior to S/G dryout** helps ensure RCS pressure prevents injecting water into the RCS.

B. **Incorrect** - EOP-3 directs OTCC initiation when **BOTH** S/G's <-350 inches.

C. **Incorrect** - EOP-3 directs OTCC initiation when **BOTH** S/G's <-350 inches. EOP-3 Basis states that initiating OTCC **prior to S/G dryout** helps ensure RCS pressure prevents injecting water into the RCS.

D. **Incorrect** - EOP-3 directs OTCC initiation when **BOTH** S/G's <-350 inches.

Question 11 (Q97135)			
Topic:	Use of PORVs during a LOAF		
Tier/Group:	1/1		
K/A Info:	054 - Loss of Main Feedwater • AK3 - Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater: • AK3.05 - HPI/PORV cycling upon total feedwater loss		
RO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EOP-3, Loss of All Feedwater; EOP-3, Loss of All Feedwater Technical Basis Document		
Comments:	None		

12. 055 - Station Blackout

During an extended station blackout event what are the fail positions of the following valves:

- Atmospheric Dump Valves
- Containment Spray CV's
- Letdown Isolation CV's

A. Close, Open, Close

B. Open, Open, Open

C. Open, Close, Open

D. Close, Close, Close

Answer: A

Answer Explanation:

A. **Correct** - Fail position of ADV's is shut to prevent excessive cooldown, Fail position of Containment Spray CV's is open to ensure spray flow on loss of air, Fail position of Letdown CV's is close to minimize loss of inventory during an accident concurrent with loss of air.

B. **Incorrect** - ADVs and L/D CVs fail closed.

C. **Incorrect** - Fail position of ADV's is shut to prevent excessive cooldown, Fail position of Containment Spray CV's is open to ensure spray flow on loss of air, Fail position of Letdown CV's is close to minimize loss of inventory during an accident concurrent with loss of air.

D. **Incorrect** - Fail position of Containment Spray CV's is open to ensure spray flow on loss of air.

Students are not expected to memorize the fail position of all air operated valves, therefore this question requires the students to process the purpose of each of these sets of valves and determine which way they should fail to keep plant safe that is reason behind the higher cognitive classification of this question.

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Question 12 (Q97136)			
Topic:	Extended SBO I/A Effects		
Tier/Group:	1/1		
K/A Info:	055 - Station Blackout • EA2 - Ability to determine or interpret the following as they apply to a Station Blackout: • EA2.01 - Existing valve positioning on a loss of instrument air system		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OM-35, Unit-1 Main & Reheat Steam OM-73, Unit-1 Chemical & Volume Control System OM-74, Unit-1 Safety Injection & Containment Spray		
Comments:	None		

13. 056 - Loss of Off-Site Power

Given both Units operating at 100% power:

One second following a complete loss of offsite power what is the status of Reactor Coolant Pumps on both units?

A. U-1 RCPs are running, U-2 RCPs are off

B. U-1 RCPs are off, U-2 RCPs are running

C. All RCPs on both Units, are off

D. All RCPs on both Units, are running

Answer: D

Answer Explanation:

A. **Incorrect** - All on due to turbine trip circuit delay on U-1 and coast down circuit on U-2

B. **Incorrect** - All on due to turbine trip circuit delay on U-1 and coast down circuit on U-2

C. **Incorrect** - All on due to turbine trip circuit delay on U-1 and coast down circuit on U-2

D. **Correct** - Unit-1: If the Turbine was paralleled to the grid, the Turbine Generator output breakers are verified open. The breakers are verified open (which directs the operator to open the breakers if they are closed) to protect against a loss of 11 125 VDC Bus, which will prevent the breakers from receiving an open signal. Following a turbine trip, the generator protection circuitry will detect reverse power, causing the Turbine Generator output breakers to trip. This trip will not occur during a turbine warm-up prior to paralleling the Generator to the grid, since the protection circuitry will not detect a reverse power condition. If offsite power is lost, the protection circuitry will not detect a reverse power condition. If the Turbine was paralleled to the grid, plant loads will slow the Turbine Generator until generator underfrequency opens breaker 252-1104, which ultimately deenergizes the generator voltage regulator. The Turbine Generator output breakers will stay closed until generator volts/hertz or another signal trips the breakers. This has been estimated to take 12 to 17 seconds. If an output breaker does not open following receipt of a trip signal, a protective circuit is energized to open breakers on the 500KV Ring Bus on either side of the Turbine Generator output which is still aligned.

Unit-2: - If the Turbine was paralleled to the grid, the Turbine Generator output breakers are checked open following a turbine trip, limit switches in the coastdown initiation circuit close on reverse power causing the generator output breakers to trip. This trip will not occur during a turbine warmup prior to paralleling the Generator to the grid, since the protection circuitry will not detect a reverse power condition. If an output breaker does not open following receipt of this trip signal, a protective circuit is energized to prevent turbine and generator damage from resulting from the reversed power condition. This circuit opens breakers on the 500KV Ring Bus on either side of the turbine generator output that is still aligned. It also opens the 13KV breakers powered from the bus causing a loss of power to the RCPs. If coastdown is initiated the generator output breaker will remain closed until generator voltage degrades to 80% or 20 seconds have elapsed. If coastdown is not initiated or the 20 seconds have elapsed, the operator is directed to open the breakers. This action protects against a loss of 21 125 VDC Bus, which will prevent the breakers from receiving an open signal.

Question 13 (Q97137)			
Topic:	RCP status on LOOP		
Tier/Group:	1/1		
K/A Info:	056 - Loss of Off-site Power • 2.2 - Equipment Control • 2.2.3 - Knowledge of the design, procedural, and operational differences between units.		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EOP-0, Post-Trip Immediate Actions Technical Basis		
Comments:	None		

14. 057 - Loss of Vital AC Inst. Bus

U-1 reactor tripped with fault on 11 4KV bus. During EOP-0 the CRS directs tying 1Y09 and 1Y10.

Which **ONE** of the following functions is restored and what is its purpose?

A. VCT Level switch is re-energized;
Allows realigning Charging Pump suction to VCT

B. Pressurizer level switch is re-energized;
Allows Charging Pumps to cycle off/on to control Pressurizer level

C. ADV/TBV controllers are re-energized;
Allows RCS temperature control from the Control Room

D. PORV/Safety Valve acoustic monitor flow indications are re-energized;
Allows diagnosis of PORV/Safety Valve leakage

Answer: C

Answer Explanation:

A. **Incorrect** - VCT level switch powered from 1Y10

B. **Incorrect** - PZR level switch powered from 1Y10

C. **Correct** - ADV/TBV controllers are powered from 1Y09 which was lost in stem of question

D. **Incorrect** - Acoustic monitors are powered from 1Y10

Question 14 (Q97152)			
Topic:	Knowledge and purpose components Loss of Vital AC		
Tier/Group:	1/1		
K/A Info:	057 - Loss of Vital AC Inst. Bus • 2.1 - Conduct of Operations • 2.1.28 - Knowledge of the purpose and function of major system components and controls.		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power		
Comments:	None		

15. 058 - Loss of DC Power

Given the following:

- Both Units are operating at 100% power when a Station Blackout occurs.
- 11, 12, 21, and 22 125V DC Bus voltages are approaching 105V DC

Which **ONE** of the following Diesel Generator combinations, if any, must be restored to prevent eventual transition to EOP-8?

A. None

B. 1A; 2A

C. 1A; 0C on 14 4KV bus

D. 1B; 0C on 24 4KV bus

Answer: C

Answer Explanation:

A. **Incorrect** - A Battery Charger must be energized on each of the four 125V DC Vital buses to maintain Bus voltages in the acceptable range. EOP-7 SFSC requires ALL 125V DC Vital Bus voltages are maintained >105V.

B. **Incorrect** - The combination of two "A" D/Gs leaves 12 and 21 125V DC Buses without Battery Chargers. 125V DC Bus voltages for these two Buses would eventually lower to < 105V necessitating transition to EOP-8 due to no longer meeting the SFSC

C. **Correct** - The combination of any "A" D/G and the OC D/G on 14 or 24 4KV Bus provides power to a Battery Charger on each of the 125V DC Buses. 125V DC Bus voltages for all Buses would increase towards nominal values. Transition to EOP-8 due to no longer meeting the SFSC

D. **Incorrect** - The combination of any "B" D/G and the OC D/G on 14 or 24 4KV Bus leaves 11 and 22 125V DC Buses without Battery Chargers. 125V DC Bus voltages for these two Buses would eventually lower to < 105V necessitating transition to EOP-8 due to no longer meeting the SFSC

Question 15 (Q97161)			
Topic:	Loss of DC / Battery Chargers		
Tier/Group:	1/1		
K/A Info:	2.8		
RO Importance:	058 - Loss of DC Power • AK1 - Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: • AK1.01 - Battery charger equipment and instrumentation		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(8)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - modified question		
Technical references:	OI-27D-2, Station Power 480 Volt System Breaker Lineup		
Comments:	Modified from Q92882		

16. 065 - Loss of Instrument Air

Under which **ONE** of the following conditions would manual operation of the Instrument Air Compressors Total Closure valves be **required**?

A. Removing compressor(s) from service (tagging).

B. Compressor operation during a loss of control power to the Unloader valve.

C. Manually loading the compressor(s) following a total Loss of Instrument Air.

D. Operation of a compressor while coming up to normal speed prior to being loaded.

Answer: C

Answer Explanation:

A. **Incorrect** - Instrument Air Header pressure is unaffected by simply removing an I/A Compressor from service. Therefore operation of the total closure valve is not required.

B. **Incorrect** - OI-19 states: The Total Closure Valve must be operated manually when receiver pressure is less than 30 PSIG. The condition stated does not result in a low I/A header pressure. Therefore operation of the total closure valve is not required.

C. **Correct** - OI-19 states: The Total Closure Valve must be operated manually when receiver pressure is less than 30 PSIG.

D. **Incorrect** - OI-19 states: The Total Closure Valve must be operated manually when receiver pressure is less than 30 PSIG. The condition stated does not result in a low I/A header pressure. Therefore operation of the total closure valve is not required.

Question 16 (Q41808)			
Topic:	Explain the S/U procedure for an Instrument Air Compressor/apply the general precautions		
Tier/Group:	1/1		
K/A Info:	065 - Loss of Instrument Air • AA1- Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: • AA1.01- Remote Manual Loaders		
RO Importance:	2.7*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	SD-019, Compressed Air OI-19, Instrument Air AOP-7D, Loss of Instrument Air		
Comments:	None		

17. 077 - Generator Voltage and Electric Grid Disturbances

Unit 1 Main Generator is currently carrying 200 MVARS - LAGGING. Unit 2 Main Generator is shutdown. A Major grid disturbance has occurred and the System Operator - Bulk Power (SO-BP) requests that Calvert Cliffs raise 500KV line voltage from 518 to 521 KV.

Which **ONE** of the following statements is true regarding the effect of this adjustment on MVARS being carried?

A. MVARS will be unaffected

B. MVARS will change only if MWe changes

C. MVARS will DECREASE to the LEAD direction as line voltage increases

D. MVARS will INCREASE in the LAG direction as line voltage increases

Answer: D

Answer Explanation:

A. **Incorrect** – Since VARS are currently in lag direction raising voltage will make them raise further in lag direction

B. **Incorrect** - VARS change based on voltage changes, a change in actual load is not required to change VARS

C. **Incorrect** – Since VARS are currently in lag direction raising voltage will make them raise further in lag direction

D. **Correct** – Raising voltage will cause lagging VARS to raise further in the lag direction.

Question 17 (Q24852)			
Topic:	Turbine Generator controls effect on VARS during grid disturbance		
Tier/Group:	1/1		
K/A Info:	077 - Generator Voltage and Electric Grid Disturbances • AA1 - Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: • AA1.02 - Turbine / generator controls		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-7M, Major Grid Disturbances		
Comments:	None		

18. CE/E05 - Excess Steam Demand

The following conditions exist on Unit-1 following a Refueling Outage Startup:

- Mode 2 with power being held at 4%
- 11 SGFP in service on Main Steam
- Pressurizer Pressure 2250 PSIA
- Performing Shell warming on the Main Turbine
- An overcooling event occurs and T_{COLD} has reached 521°F and continues to lower slowly
- The appropriate procedure has been implemented

Which **ONE** of the following sets of actions are procedurally directed to prevent T_{COLD} from continuing to lower?

A. Trip the reactor, shut the MSIVs and upstream drains, and initiate auxiliary feedwater.

B. Shut the SG blowdown valves, shut the MSIV upstream drains, and shift steam seals to auxiliary steam.

C. Secure main turbine shell warming, shift steam seals to auxiliary steam and initiate auxiliary feedwater.

D. Shut the MSIVs and upstream drains, reduce reactor power to maintain between 2 to 4%, and initiate auxiliary feed.

Answer: B

Answer Explanation:

A. **Incorrect** – Trip criteria in AOP-7K is RCS temp <515°, therefore tripping reactor at 521°F is incorrect

B. **Correct** - These actions are specified in AOP-7K, Step IV.D.4

C. **Incorrect** – AOP-7K directs tripping turbine vice just securing shell warming and initiating auxiliary feed at existing power level is incorrect as well (power must be lowered <1%)

D. **Incorrect** – Lowering power to 1% is an action called out in AOP-7K Step IV.D.4, not 2-4%

Question 18 (Q28505)			
Topic:	Determine the appropriate actions for overcooling event in Mode 2		
Tier/Group:	1/1		
K/A Info:	CE/E05 - Excess Steam Demand • EK2 - Knowledge of the interrelations between the (Excess Steam Demand) • EK2.2 - Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operations of these systems to the operation of the facility.		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-7K, Overcooling Event In Mode One Or Two		
Comments:	None		

19. 032 - Loss of Source Range NI

U-2 is preparing to parallel the turbine generator to the grid following a MOC trip recovery. Channel B Wide Range Nuclear Instrument fails low due to a High Voltage Power Supply problem.

Prior to any operator actions being performed which **ONE** of the statements below is correct regarding RPS trip logic?

A. 1 out of 3 logic condition for SUR Trip

B. 2 out of 3 logic condition for SUR Trip

C. 2 out of 4 logic condition for SUR Trip

D. SUR Trip bypassed on all 4 channels of RPS

Answer: B

Answer Explanation:

A. **Incorrect** – HV PS failure will fail WRNI low, this will bypass the SUR trip on Channel B. Power is <14% so other 3 channels are not bypassed. The distractor is plausible because this answer would be correct if the PS failure caused the T/U to trip

B. **Correct** - HV PS failure will fail WRNI low, this will bypass the SUR trip on Channel B. Power is <14% so other 3 channels are not bypassed. Therefore 2 of 3 logic is correct.

C. **Incorrect** - HV PS failure will fail WRNI low, this will bypass the SUR trip on Channel B. Power is <14% so other 3 channels are not bypassed. This distractor is plausible if student did not remember that low WRNI reading also bypasses SUR trip.

D. **Incorrect** - HV PS failure will fail WRNI low, this will bypass the SUR trip on Channel B. Power is <14% so other 3 channels are not bypassed. This answer is plausible if student does not recognize Reactor Power is <14% (Parallel turbine is below 14%)

Question 19 (Q97162)			
Topic:	Relationship between WRNI & power supplies		
Tier/Group:	1/2		
K/A Info:	032 - Loss of Source Range NI • AK2 - Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: • AK2.01 – Power supplies, including proper switch positions		
RO Importance:	2.7*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	SD-058, Reactor Protective System Description		
Comments:	None		

20. 036 - Fuel Handling Accident

U-2 is in mode 6 performing a fuel shuffle. A core-to-core fuel move is in progress. While withdrawing a fuel assembly, Chemistry informs the Control Room that boron sample has lowered by 50 ppm unexpectedly.

Which **ONE** of the following actions is required by AOP-1A "Inadvertent Boron Dilution" with respect to the fuel assembly being moved?

A. Complete current fuel move

B. Leave fuel assembly in its current position

C. Reinsert the fuel assembly into its original location

D. Complete withdrawing fuel assembly and place it in the RFM upender

Answer: D

Answer Explanation:

A. **Incorrect** - AOP-1A, specifically directs placing the fuel assembly in the RFP upender or and approved SFP location. Completing current move would have you reinsert assembly into core which is incorrect due to SDM concerns.

B. **Incorrect** - AOP-1A, specifically directs placing the fuel assembly in the RFP upender or and approved SFP location. Leaving assembly partially inserted into core is incorrect due to SDM concerns.

C. **Incorrect** - AOP-1A, specifically directs placing the fuel assembly in the RFP upender or and approved SFP location. Reinserting fuel assembly back in original location is incorrect due to SDM concerns

D. **Correct** - AOP-1A, specifically directs placing the fuel assembly in the RFP upender or and approved SFP location.

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Question 20 (Q97163)			
Topic:	Maintaining SDM during a Fuel handling incident		
Tier/Group:	1/2		
K/A Info:	036 - Fuel Handling Accident • AK1 - Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents • AK1.02 - SDM		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	AOP-6D, Fuel Handling Incident		
Comments:	None		

21. 060 - Accidental Gaseous RadWaste Release

An uncontrolled gaseous release to the environment has been detected. AOP-6C "Accidental Gaseous Waste Release" has been implemented.

Which valve(s) are directed to be verified shut in this condition?

A. WGDT Outlet valves

B. Purification IX inlet valves

C. PS-5464-CV RCS Sample valve

D. Letdown Filter Inlet and Outlet valves

Answer: A

Answer Explanation:

All valves listed are in AOP-6C but only the WGDT valves are in the section for uncontrolled release to the environment. Distractors B,C & D are all in the High Airborne in the Aux Bldg section. Upper cognitive because student must determine which valves could result in release to environment.

A. **Correct** - See above

B. **Incorrect** - See above

C. **Incorrect** - See above

D. **Incorrect** - See above

Question 21 (Q97164)			
Topic:	Accidental Gaseous Release valve lineup		
Tier/Group:	1/2		
K/A Info:	060 - Accidental Gaseous RadWaste Release • AA2 - Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: • AA2.06 - Valve lineup for release of radioactive gases		
RO Importance:	3.6*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	AOP-6C, Accidental Gaseous Waste Release		
Comments:	None		

22. 061 - ARM System Alarms

The Wide Range Noble Gas Monitor (RIC-5415) would provide early detection of radioactivity during which **ONE** of the following events?

A. 40 GPM Leak on the VCT

B. 40 GPM RCS cold leg leak

C. 40 GPM Letdown HX to CC leak

D. 40 GPM Spent Fuel Pool HX to SRW leak

Answer: A

Answer Explanation:

A. **Correct** - A leak on the VCT would be monitored by the WRNGM since the Auxiliary Building Ventilation System discharges thru the WRNGM

B. **Incorrect** - An RCS Cold Leg leak would be confined to the Containment, therefore not monitored by the WRNGM.

C. **Incorrect** - A LDHX tube leak would be monitored by the CCW RMS, therefore not monitored by the WRNGM early on. Eventually the WRNGM may detect radioactivity as a result of the CC Head tank being vented to the Auxiliary Building Atmosphere.

D. **Incorrect** - An SFP HX leak would be monitored by the SRW RMS, therefore not monitored by the WRNGM. Eventually the WRNGM may detect radioactivity as a result of the SRW Head tank being vented to the Auxiliary Building Atmosphere.

Question 22 (Q24708)			
Topic:	The Wide Range Noble Gas Monitor (RIC-5415) would detect radioactivity during which accident?		
Tier/Group:	1/2		
K/A Info:	061 - ARM System Alarms • 2.1 - Conduct of Operations • 2.1.27 - Knowledge of system purpose and/or function		
RO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	OI-35, Radiation Monitoring System SD-077/079, Radiation Monitoring System		
Comments:	None		

23. 067 - Plant Fire On-site

A severe fire has occurred in the Control Room.

Which **ONE** of the following groups of systems/components can be electrically **isolated** at the Remote Shutdown Panel (1/2C43) to isolate the fire effects?

A. Letdown, RCS Sampling, Controlled Bleed-Off, S/G Blowdown

B. Letdown, Controlled Bleed-Off, Atmospheric Dump Valves, S/G Blowdown

C. Controlled Bleed-Off, RCS Sampling, Auxiliary Feed Water, S/G Blowdown

D. RCS Sampling, Atmospheric Dump Valves, S/G Blowdown, Auxiliary Feed Water

Answer: A

Answer Explanation:

A. **Correct** - Per AOP-9A these components are isolated from the effects of the fire through operator actions taken at the Remote Shutdown Panel

B. **Incorrect** - Per AOP-9A these components, **with the exception of ADVs**, are isolated from the effects of the fire through operator actions taken at the Remote Shutdown Panel

C. **Incorrect** - Per AOP-9A these components, **with the exception of Auxiliary Feedwater and the SRW HXs**, are isolated from the effects of the fire through operator actions taken at the Remote Shutdown Panel

D. **Incorrect** - Per AOP-9A these components, **with the exception of the ADVs and Auxiliary Feedwater**, are isolated from the effects of the fire through operator actions taken at the Remote Shutdown Panel

Question 23 (Q97140)			
Topic:	Fire Suppression Activities		
Tier/Group:	1/2		
K/A Info:	067 - Plant Fire On-site • AA2 - Ability to determine and interpret the following as they apply to the Plant Fire on Site: • AA2.14 - Equipment that will be affected by fire suppression activities in each zone		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	AOP-9A - Control Room Evacuation And Safe Shutdown Due To A Severe Control Room Fire		
Comments:	None		

24. 068 - Control Room Evacuation

A severe fire has occurred in the Control Room concurrent with a loss of offsite power. Which buses are energized and from what source of power using the appropriate procedure?

A. 14 4KV bus from the 0C D/G;
   21 4KV bus from the 2A D/G

B. 11 4 KV bus from the 1A D/G;
   24 4KV bus from the 0C D/G

C. 11 4KV bus from the 0C D/G;
   24 4KV bus from the 0C D/G

D. 14 4KV bus from the 0C D/G;
   21 4KV bus from the 0C D/G

Answer: C

Answer Explanation:

A. **Incorrect** - AOP-9A-1/2 directs aligning the OC D/G to 4KV Buses 11 & 24

B. **Incorrect** - AOP-9A-1/2 directs aligning the OC D/G to 4KV Buses 11 & 24

C. **Correct** - AOP-9A-1/2 directs aligning the OC D/G to 4KV Buses 11 & 24

D. **Incorrect** - AOP-9A-1/2 directs aligning the OC D/G to 4KV Buses 11 & 24

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Question 24 (Q97141)			
Topic:	4KV Busses powered during a CR Evacuation		
Tier/Group:	1/2		
K/A Info:	068 - Control Room Evacuation • AK2 - Knowledge of the interrelations between the Control Room Evacuation and the following: • AK2.07 - ED/G		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	AOP-9A - Control Room Evacuation And Safe Shutdown Due To A Severe Control Room Fire		
Comments:	None		

25. Loss of Containment Barrier per ERPIP

Which **ONE** of the following conditions would be considered a loss/potential loss of the containment barrier per the Emergency Action Level Threshold for classification?

A. CET temp reaches 700˚F

B. Containment hydrogen concentration of 8%

C. S/G tube leak with a loss of condenser vacuum

D. Containment pressure peaks at 47 PSIG and slowly lowers

Answer: B

Answer Explanation:

A. **Incorrect** – Per ERPIP 3.0 (EAL-Hot), Fission Product Barrier Matrix, the criteria for a Potential Loss of the Containment Barrier is > 1200°F for > 15 minutes

B. **Correct** - Per ERPIP 3.0 (EAL-Hot), Fission Product Barrier Matrix, the criteria for a Potential Loss of the Containment Barrier is Hydrogen concentration ≥ 4%

C. **Incorrect** - Per ERPIP 3.0 (EAL-Hot), Fission Product Barrier Matrix, the criteria for a Potential Loss of the Containment Barrier is that the ruptured S/G is also faulted

D. **Incorrect** - Per ERPIP 3.0 (EAL-Hot), Fission Product Barrier Matrix, the criteria for a Potential Loss of the Containment Barrier is a Containment pressure rise that is followed by a rapid unexplained depressurization

RO's are not expected to memorize the EAL chart but they are expected to analyze each of the given conditions and determine which one challenges the containment as a fission product barrier, therefore higher cognitive.

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Question 25 (Q97142)			
Topic:	Loss of Containment Barrier per ERPIP		
Tier/Group:	1/2		
K/A Info:	069 - Loss of Containment Integrity • 2.4 - Emergency Procedures / Plan • 2.4.41 - Knowledge of the emergency action level thresholds and classifications		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	ERPIP 3.0, Emergency Response Plan Implementation Procedure		
Comments:	None		

26. 076 - High Reactor Coolant Activity

Using Provided Reference(s)

U-1 is currently escalating power to 100% following a Refueling Outage. The following conditions exist:

- The unit is currently at 50% reactor power
- "RAD MON LVL HI" alarm at 1C07 annunciated and chemistry has validated the alarm
- AOP-6A has been implemented in response to high RCS Activity.
- Chemistry reports the following values:
  - Dose Equivalent I-131 = 82 uCi/gm
  - Gross Activity = 115 uCi/gm
  - E-Bar = 0.805

Based on these values, which **ONE** of the following sets of statements is true?

A. Gross Activity is acceptable;
Dose Equivalent I-131 is in the unacceptable region of the figure

B. Gross Activity is acceptable;
Dose Equivalent I-131 is high in the acceptable region

C. Gross Activity is high;
Dose Equivalent I-131 is in the unacceptable region of the figure

D. Gross Activity is high;
Dose Equivalent I-131 is high in the acceptable region

Answer: B

Answer Explanation

A. **Incorrect** - Per TS 3.4.15 the gross activity is SAT since 100/E-bar limit is 124 μCi/gm and **specific activity limit is 90 μCi/gm I-131**

B. **Correct** - Per TS 3.4.15 the gross activity is SAT since 100/E-bar limit is 124 μCi/gm and specific activity limit is 90 μCi/gm I-131

C. **Incorrect** - Per TS 3.4.15 the **gross activity is SAT since 100/E-bar limit is 124 μCi/gm and specific activity limit is 90 μCi/gm I-131**

D. **Incorrect** – Per TS 3.4.15 the **gross activity is SAT since 100/E-bar limit is 124 μCi/gm** and specific activity limit is 90 μCi/gm I-131

Question 26 (Q51251)			
Topic:	Recall parameters monitored for entry into AOP 6A		
Tier/Group:	1/2		
K/A Info:	076 - High Reactor Coolant Activity • AA1 - Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: • AA1.04 - Failed fuel-monitoring equipment		
RO Importance:	3.2		
Proposed references to be provided to applicant:	T.S. 3.4.15 and associated Bases		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-6A, Abnormal Reactor Coolant Chemistry/Activity T.S. 3.4.15 and associated Bases		
Comments:	None		

27. CE/A11 - RCS Overcooling

U-1 is returning to full power operation following Main Turbine Valve Testing, when the following occurred:

- Turbine Bypass Valve, 1-MS-3940-CV failed open.
- AOP-7K, Overcooling Event, has been implemented.

Which **ONE** of the following sets of actions/reasons is appropriate, to initially stabilize the plant, for the stated conditions?

A. Turbine load is lowered, and CEA's are withdrawn to restore T_{COLD} to program

B. The TBV is manually isolated to restore T_{COLD} to program

C. Turbine load is lowered to restore T_{COLD} to program

D. Withdraw CEA's to restore T_{COLD} to program

Answer: C

Answer Explanation:

A. **Incorrect** - CEA withdrawal is not in accordance with the direction provided for stabilizing the plant. CEAs would be used to adjust reactor power, if necessary, once RCS temperature has been stabilized if power was continuing to lower.

B. **Incorrect** - Manual isolation of the affected TBV is an action performed, in a controlled sequence, **ONCE** the plant has been stabilized.

C. **Correct** - Per AOP-7K, restoring T_{COLD} to its program value is the appropriate actions for placing the plant in a stable condition. The reasons stated for the action is correct.

D. **Incorrect** - CEA withdrawal is not in accordance with the direction provided for stabilizing the plant. CEAs would be used to adjust reactor power, if necessary, once RCS temperature has been stabilized if power was continuing to lower

Question 27 (Q51252)			
Topic:	Response to an Overcooling Event		
Tier/Group:	1/2		
K/A Info:	CE/A11 - RCS Overcooling • AK3 - Knowledge of the reasons for the following responses as they apply to the (RCS Overcooling) • AK3.3 - Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-7K, Overcooling Event In Mode One Or Two		
Comments:	None		

28. 003 - Reactor Coolant Pump System

Which **ONE** of the following describes **ALL** of the interlocks that must be satisfied to start a Reactor Coolant Pump?

A. Oil lift pressure > 600 PSI, synchronizing stick inserted

B. Oil lift pressure > 600 PSI, sufficient CCW flow exists

C. Sufficient CCW flow exists, synchronizing stick inserted

D. Oil lift pressure > 600 PSI, sufficient CCW flow exists, synchronizing stick inserted

Answer: D

Answer Explanation:

A. **Incorrect** - Additionally, the CCW flow switches must be made up.

B. **Incorrect** - The Synchronizing stick must be inserted.

C. **Incorrect** - The Oil lift pump must be running for initial start of the RCP.

D. **Correct** - The Synchronizing stick must be inserted and the CCW flow switch must be made up. The Oil lift pump must be running for initial start of the RCP.

Question 28 (Q92714)			
Topic:	RCP Start Interlocks		
Tier/Group:	2/1		
K/A Info:	003 - Reactor Coolant Pump System • K4 - Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: • K4.04 – Adequate cooling of RCP motor and seals		
RO Importance:	2.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	OI-1A, Reactor Coolant System and Pump Operations		
Comments:	None		

29. 003 - Reactor Coolant Pump System

U-2 tripped due to a SGFP coupling problem, and being held in Mode 3 awaiting repairs.

Which **ONE** of the sets of conditions below is the minimum required to meet RCS Coolant Loop Operability Verification requirements?

A. 2 RCP's operating in each loop with both S/G levels >-40 inches

B. 1 RCP operating in each loop with both S/G levels >-40 inches

C. 2 RCP's operating in one loop with associated S/G level >-40 inches

D. 1 RCP operating in one loop with associated S/G level >-40 inches

Answer: B

Answer Explanation:

A. **Incorrect** - STP-O-94A, Coolant Loop Operability Verification (Mode 3) requires an RCP operating in each loop with both S/G levels >-40 inches

B. **Correct** - STP-O-94A, Coolant Loop Operability Verification (Mode 3) requires an RCP operating in each loop with both S/G levels >-40 inches

C. **Incorrect** - STP-O-94A, Coolant Loop Operability Verification (Mode 3) requires an RCP operating in each loop with both S/G levels >-40 inches

D. **Incorrect** - STP-O-94A, Coolant Loop Operability Verification (Mode 3) requires an RCP operating in each loop with both S/G levels >-40 inches

Question 29 (Q97143)			
Topic:	Knowledge of RCP surveillance procedures		
Tier/Group:	2/1		
K/A Info:	003 - Reactor Coolant Pump System • 2.1 - Equipment Control • 2.1.2 - Knowledge of surveillance procedures		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	STP-O-94A, Coolant Loop Operability Verification (Mode 3)		
Comments:	None		

30. 004 - Chemical and Volume Control

When collapsing the PZR bubble with RCS solid, which **ONE** of the following sets of (1) abnormal conditions and (2) consequences due to this abnormal condition would be true?

A. (1) Backpressure Regulator setpoint fails low;
(2) RCS Pressure lowers

B. (1) Backpressure Regulator setpoint fails low;
(2) RCS Pressure rises

C. (1) Running charging pump shaft shears;
(2) Letdown Control Valve shuts

D. (1) Running charging pump shaft shears;
(2) Letdown Control Valve opens

Answer: A

Answer Explanation:

A. **Correct** - If Backpressure regulator setpoint fails low the backpressure is higher than setpoint so backpressure control valves open to lower pressure to setpoint.

B. **Incorrect** - If Backpressure regulator setpoint fails low then the backpressure is higher than setpoint so backpressure control valves open to **lower** pressure to setpoint.

C. **Incorrect** - The Letdown control valves are in the manual mode of control and are full open with the plant in this condition. No control function occurs

D. **Incorrect** - The Letdown control valves are in the manual mode of control and are full open with the plant in this condition. No control function occurs

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Question 30 (Q97144)			
Topic:	RCS Pressure control using CVCS		
Tier/Group:	2/1		
K/A Info:	004 - Chemical and Volume Control • K6 - Knowledge of the effect of a loss or malfunction on the following CVCS components: • K6.26 - Methods of pressure control of solid plant (PZR relief and water inventory)		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OP-7, Shutdown Operations		
Comments:	None		

31. 004 - Chemical and Volume Control

A Loss of Coolant Accident has occurred on U-2 which resulted in a Containment peak pressure of 5 PSIG. The CRS has directed that you lower subcooling to the lower end of the band.

Which **ONE** of the following actions is required by EOP-5 "Loss of Coolant Accident" under these conditions?

A. Open Aux Spray Control Valve, 2-CVC-517-CV;
Operate Charging Header Loop Stops 2-CVC-518 & 519-CV's as necessary

B. Place the Pressurizer Spray Valve Controller, 2-HIC-100, to manual;
Raise the output to open the main spray valve to desired amount

C. Open the Atmospheric Steam Dump Valves by placing 2-HIC-4056, in manual;
Operate the ADV's by adjusting controller output to obtain the desired result

D. Place Turbine Bypass Valve Controller, 2-PIC-4056, in manual;
Lower the output to obtain the desired result

Answer: A

Answer Explanation:

A. **Correct** - Per EOP-5, Auxiliary Spray will be initiated by opening CVC-517 and shutting CVC-518 & 519.

B. **Incorrect** - Raising the output of 2-HIC-100 would normally result in lowering subcooling **IF** the RCPs were running. RCPs would have been secured when CIS actuated at 2.8 PSIG.

C. **Incorrect** - Raising the output of the ADV controller would lower RCS temperature resulting in an increase in RCS subcooling

D. **Incorrect** - For the conditions stated CSAS will have actuated, closing the MSIVs. The TBVs will not be available for control of RCS subcooling

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Question 31 (Q97165)			
Topic:	Actions to control RCS subcooling		
Tier/Group:	2/1		
K/A Info:	004 - Chemical and Volume Control • A4 - Ability to manually operate and/or monitor in the control room: • A4.09 - PZR spray and heater controls		
RO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EOP-5, Loss of Coolant Accident		
Comments:	None		

32. 005 - Residual Heat Removal

Which **ONE** of the following groups of instruments are designed for Post-Accident Monitoring?

1. S/G Narrow Range level, LI-1111/1121
2. PZR Low Range Pressure, PI-103/103-1
3. Containment Narrow Range Pressure, PI-5308
4. PZR Level, LI-110X/110Y

A. 1, 2, 3

B. 1, 3, 4

C. 1, 2, 4

D. 2, 3, 4

Answer: D

Answer Explanation:

A. **Incorrect** – All instruments listed **except** S/G Narrow Range level are PAMS

B. **Incorrect** - All instruments listed **except** S/G Narrow Range level are PAMS

C. **Incorrect** - All instruments listed **except** S/G Narrow Range level are PAMS

D. **Correct** - All instruments listed **except** S/G Narrow Range level are PAMS

Question 32 (Q97145)			
Topic:	RHR Post Accident Instrumentation		
Tier/Group:	2/1		
K/A Info:	005 - Residual Heat Removal • 2.4 - Emergency Procedures / Plan • 2.4.3 - Ability to identify post-accident instrumentation		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(6)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:			
Comments:	None		

33. 005 - Residual Heat Removal

Which **ONE** of the following describes (1) adequate mixing flow with SDC in service; and (2) the basis for this requirement?

A. (1) 3000 GPM SDC flow with PZR manway removed
(2) Prevents dilute pockets in the RCS

B. (1) 1500 GPM SDC flow with PZR manway removed
(2) Maintains adequate Shutdown Margin

C. (1) 3000 GPM SDC flow with S/G tubes full
(2) Prevents dilute pockets in the RCS

D. (1) 1500 GPM SDC flow with S/G tubes full
(2) Maintains adequate Shutdown Margin

Answer: C

Answer Explanation:

OI-2B and OP-7 define adequate mixing as:

"At least 3000 GPM flow through the core **AND** at least 500 GPM flow through both S/Gs." 500 GPM through both S/G's is not assured without a bubble in the PZR. The basis for these limits is to prevent dilute pockets.

A. **Incorrect** - With PZR manway removed there is no chance of a bubble in the PZR

B. **Incorrect** - 1500 GPM does not meet minimum flow and maintaining SDM is not the basis for requirement

C. **Correct - 3000 GPM meets the limit and S/G tubes full also meets requirement and basis is to prevent dilute pockets**

D. **Incorrect** - 1500 GPM does not meet minimum flow and maintaining SDM is not the basis for requirement

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Question 33 (Q97166)			
Topic:	Conditions permitting dilution of the RCS		
Tier/Group:	2/1		
K/A Info:	005 - Residual Heat Removal • K5 - Knowledge of the operational implications of the following concepts as they apply the RHRS: • K5.03 - Reactivity effects of RHR fill water		
RO Importance:	2.9*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OI-2B, CVCS Boration, Dilution And Makeup Operations OP-7, Shutdown Operations		
Comments:	None		

34. 006 - Emergency Core Cooling

U-1 is operating at 100% power when an invalid SIAS “A” actuates.

Which **ONE** of the following statements describe components operated and how SIAS is reset with minimal impact on plant operation?

A. Aux HPSI Header MOV’s open;
Match all handswitches on the SIAS verification checklist and depress the SIAS reset pushbutton on 1C10

B. All SRW Turbine Bldg Header Isolation CV’s shut;
Match all handswitches on the SIAS verification checklist and depress the SIAS reset pushbutton on 1C10

C. Aux HPSI Header MOV’s open;
Match asterisked handswitches on the SIAS verification checklist and depress the SIAS reset pushbutton on 1C10

D. All SRW Turbine Bldg Header Isolation CV’s shut;
Match asterisked handswitches on the SIAS verification checklist and depress the SIAS reset pushbutton on 1C10

Answer: C

Answer Explanation:

A. **Incorrect** - The Aux HPSI Hdr MOVs open on SIAS “A”, however **only** the handswitches annotated by an asterisk in Attachment (2) of the EOP Attachments must be matched to reset SIAS from 1C10. Matching all handswitches on the SIAS verification checklist would essentially initiate a complete SIAS “A” & “B” having a significant effect on power operation (boration & isolation of SRW to the Turbine Building).

B. **Incorrect** – Only 2 SRW TB isolations shut on SIAS “A” not all 4, and **only** the handswitches annotated by an asterisk in Attachment (2) of the EOP Attachments must be matched to reset SIAS from 1C10

C. **Correct** - The Aux HPSI Hdr MOVs open on SIAS “**A**” and **only** the handswitches annotated by an asterisk in Attachment (2) of the EOP Attachments must be matched to reset SIAS from 1C10

D. **Incorrect** - Only 2 SRW TB isolations shut on SIAS “A” not all 4, and **only** the handswitches annotated by an asterisk in Attachment (2) of the EOP Attachments must be matched to reset SIAS from 1C10

Question 34 (Q97167)			
Topic:	Invalid SIAS impact on plant ops		
Tier/Group:	2/1		
K/A Info:	006 - Emergency Core Cooling • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.13 - Inadvertent SIS actuation		
RO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	1C08-ALM, ESFAS 11 Alarm Manual EOP Attachments, Attachment (2)		
Comments:	None		

35. 007 - Pressurizer Relief/Quench Tank

OP-7, Shutdown Operations, requires the Quench Tank be prepared for drawing a Pressurizer bubble. Current Quench Tank oxygen concentration is 5%.

What is the **maximum** oxygen concentration allowed in the Quench Tank and how is it reduced, IAW OI-1B, Quench Tank Operations?

A. 4%; Cycle QT pressure between 0.5 PSIG and 10.0 PSIG 12 times by cycling QT vent and N_2 Supply one at a time.

B. 4%; Open QT vent and throttle open N_2 Supply to maintain pressure between 0.5 PSIG and 1.5 PSIG for 15 minutes.

C. 2%; Cycle QT pressure between 0.5 PSIG and 10.0 PSIG 12 times by cycling QT vent and N_2 Supply one at a time.

D. 2%; Open QT vent and throttle open N_2 Supply to maintain pressure between 0.5 PSIG and 1.5 PSIG for 15 minutes.

Answer: B

Answer Explanation:

A. **Incorrect** - OI-1B Sect. 6.11 requirements for drawing PZR bubble state that 4% is the maximum oxygen concentration therefore C & D are incorrect. OI-1B Sect 6.10 states that to lower concentration you purge for 15 minutes vice cycling **12 times**.

B. **Correct** - OI-1B Sect. 6.11 requirements for drawing PZR bubble state that 4% is the maximum oxygen concentration. OI-1B Sect 6.10 states that to lower concentration you purge for 15 minutes.

C. **Incorrect** - OI-1B Sect. 6.11 requirements for drawing PZR bubble state that **4%** is the maximum oxygen concentration therefore C & D are incorrect. OI-1B Sect 6.10 states that to lower concentration you purge for 15 minutes vice cycling **12 times**.

D. **Incorrect** - OI-1B Sect. 6.11 requirements for drawing PZR bubble state that **4%** is the maximum oxygen concentration therefore C & D are incorrect. OI-1B Sect 6.10 states that to lower concentration you purge for 15 minutes.

Question 35 (Q97168)			
Topic:	Quench tank O_2 limit for operability		
Tier/Group:	2/1		
K/A Info:	007 - Pressurizer Relief/Quench Tank • K5 - Knowledge of the operational implications of the following concepts as the apply to PRTS: • K5.02 - Method of forming a steam bubble in the PZR		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OI-1B, Quench Tank Operations OP-7, Shutdown Operations		
Comments:	None		

36. 008 - Component Cooling Water

Which **ONE** of the following conditions require controlled restoration of Component Cooling flow to the containment following a station blackout and why?

A. Controlled Bleed-off temperature > 250°F;
RCP seals need to be rebuilt

B. Controlled Bleed-off temperature > 250°F;
Thermal shock could cause an RCS leak

C. Lower Seal temperature > 280°F;
Thermal shock could cause an RCS leak

D. Lower Seal temperature > 280°F;
RCP seals need to be rebuilt

Answer: C

Answer Explanation:

A. **Incorrect** - This condition **does** require RCP Seals be rebuilt prior to starting the affected pump, however, this condition does not prohibit the controlled restoration of CCW flow to the Containment

B. **Incorrect** - The primary concern with exceeding a CBO temperature of 250°F is that it results in the need to rebuild the seal package prior to operation of the pump.

C. **Correct** - The primary concern with exceeding a Lower Seal temperature of 280°F is that uncontrolled restoration of CCW flow could result in a thermal shock to the seals causing a seal LOCA. Controlled restoration of CCW is done to prevent thermal shock to the RCP seals and seal coolers resulting from the initiation of cooling water at elevated component temperatures

D. **Incorrect** - The primary concern with exceeding a Lower Seal temperature of 280°F is that it could result in a seal LOCA.

Question 36 (Q97146)			
Topic:	CCW Setpoints, Interlocks, Auto Actions for EOP Entry		
Tier/Group:	2/1		
K/A Info:	008 - Component Cooling Water • 2.4 - Emergency Procedures / Plan • 2.4.6 - Knowledge of EOP mitigation strategies		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EOP-7, Station Blackout, Step "IV.R"		
Comments:	None		

37. 008 - Component Cooling Water

You suspect that a Shutdown Cooling Heat Exchanger tube has ruptured while on Shutdown Cooling. RCS pressure is 100 PSIA.

Which **ONE** of the following indications could be used to confirm a Shutdown Cooling Heat Exchanger tube leak/rupture?

A. Lowering Component Cooling head tank level;
Rising PZR level;
No change in Aux Building radiation levels.

B. Rising Component Cooling head tank level;
Lowering PZR level;
Rising Aux Building radiation levels.

C. Lowering Component Cooling head tank level;
Lowering PZR level;
No change in Aux Building radiation levels.

D. Rising Component Cooling head tank level;
Rising PZR level;
Rising Aux Building radiation levels.

Answer: B

Answer Explanation:

A. **Incorrect** - RCS pressure is higher than Component Cooling Header Press (SDC HX cooling medium) which would result in RCS leakage into the CCW System. Pressurizer level would lower and Aux Bldg Radiation levels would rise

B. **Correct** - RCS pressure is higher than Component Cooling Header Press (SDC HX cooling medium) which would result in RCS leakage into the CCW System. Pressurizer level would lower and Aux Bldg Radiation levels would rise

C. **Incorrect** - RCS pressure is higher than Component Cooling Header Press (SDC HX cooling medium) which would result in RCS leakage into the CCW System. Pressurizer level would lower and Aux Bldg Radiation levels would rise

D. Incorrect - RCS pressure is higher than Component Cooling Header Press (SDC HX cooling medium) which would result in RCS leakage into the CCW System. Pressurizer level would lower and Aux Bldg Radiation levels would rise

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Question 37 (Q97183)			
Topic:	RCS Leak on a SDCHX		
Tier/Group:	2/1		
K/A Info:	008 - Component Cooling Water • A1 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: • A1.04 - Surge tank level		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-113-5-5-16		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-3B, Abnormal Shutdown Cooling Conditions, Step V.10 AOP-2A, Excessive Reactor Coolant Leakage, Step IX.5.1		
Comments:	None		

38. 010 - Pressurizer Pressure Control

Given the following:

- Unit-1 is at 100% power
- RCS Pressure Control is in AUTO
- RCS Pressure is 2250 PSIA

Which **ONE** of the following describes the **IMMEDIATE** plant response if the setpoint for the selected Pressurizer Pressure Controller fails low?

A. Spray valve controller goes to maximum output;
Proportional heaters output goes to maximum;
Actual RCS pressure will rise.

B. Spray valve controller goes to minimum output
Proportional heaters output goes to minimum
Actual RCS pressure will lower.

C. Spray valve controller goes to minimum output;
Proportional heaters output goes to maximum;
Actual RCS pressure will rise.

D. Spray valve controller goes to maximum output;
Proportional heaters output goes to minimum;
Actual RCS pressure will lower.

Answer: D

Answer Explanation:

A. **Incorrect** - Setpoint is lower than sensed press so the controller is trying to lower RCS pressure. Spray goes to maximum, **Proportional Heaters goes to minimum**, **actual RCS pressure will lower**.

B. **Incorrect** - Setpoint is lower than sensed press so the controller is trying to lower RCS pressure. **Spray goes to maximum**, Proportional Heaters goes to minimum, actual RCS pressure will lower.

C. **Incorrect** - Setpoint is lower than sensed press so the controller is trying to lower RCS pressure. **Spray goes to maximum, Proportional Heaters goes to minimum, actual RCS pressure will lower**.

D. **Correct** - Setpoint is lower than sensed press so the controller is trying to lower RCS pressure. Spray goes to maximum, Proportional Heaters goes to minimum, actual RCS pressure will lower.

Question 38 (Q97172)			
Topic:	Selected Pzr Pressure Control Channel failure		
Tier/Group:	2/1		
K/A Info:	010 - Pressurizer Pressure Control • K3 - Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: • K3.01 - RCS		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - modified question		
Technical references:	1C06-ALM, RCS Control Alarm Manual		
Comments:	Modified from Q92862		

39. 010 - Pressurizer Pressure Control

The reactor is operating at 80% power, late in core life for previous 5 days, when the following occur:

- Steam demand is lowered by 5% due to a Turbine Control Valve drifting shut.
- Pressurizer level changes by 9 inches due to transient

Which **ONE** of the following describes the **initial** response of the Pressurizer Spray valves and Pressurizer Heaters?

A. Spray valves shut, backup heaters energize.

B. Spray valves shut, backup heaters remain deenergized.

C. Spray valves open, backup heaters energize.

D. Spray valves open, backup heaters remain deenergized.

Answer: D

Answer Explanation:

A. **Incorrect** - Lowering steam demand causes Reactor Coolant System temperature to rise with a resulting increase in RCS pressure. The Pressurizer Pressure Control System responds to **deenergize** Pressurizer Heaters and **opens** the Pressurizer Spray valves to control the rising RCS pressure

B. **Incorrect** - Lowering steam demand causes Reactor Coolant System temperature to rise with a resulting increase in RCS pressure. The Pressurizer Pressure Control System responds to deenergize Pressurizer Heaters and **opens** the Pressurizer Spray valves to control the rising RCS pressure

C. **Incorrect** - Lowering steam demand causes Reactor Coolant System temperature to rise with a resulting increase in RCS pressure. The Pressurizer Pressure Control System responds to **deenergize** Pressurizer Heaters and opens the Pressurizer Spray valves to control the rising RCS pressure

D. **Correct** - Lowering steam demand causes Reactor Coolant System temperature to rise with a resulting increase in RCS pressure. The Pressurizer Pressure Control System responds to deenergize Pressurizer Heaters and opens the Pressurizer Spray valves to control the rising RCS pressure

Question 39 (Q14311)			
Topic:	Predict response of Pzr to change in steam demand		
Tier/Group:	2/1		
K/A Info:	010 - Pressurizer Pressure Control A3 - Ability to monitor automatic operation of the PZR PCS, including: A3.02 - PZR pressure		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	1C06-ALM, RCS Control Alarm Manual SD-064D, Reactor Coolant System Instrumentation System Description		
Comments:	None		

40. 012 - Reactor Protection

Which **ONE** of the following statements describes a function of the Matrix Relay Hold Pushbutton?

A. Ensures continuity from sensor to trip unit

B. Applies a test voltage to relays K-1 through K-4

C. Supplies power to test the Trip Circuit Breaker Shunt and UV coils for testing.

D. Prevents logic matrix relay contact from opening with normal power removed.

Answer: D

Answer Explanation:

A. **Incorrect** - The Matrix Relay Hold Pushbutton supplies power to maintain the individual matrix relays (AB, AC, AD, etc. …) powered to prevent trip relays K-1 thru K-4 from opening (and opening their respective TCBs) during testing.

B. **Incorrect** - The Matrix Relay Hold Pushbutton does not directly supply power to relays K-1 thru K-4.

C. **Incorrect** - The Matrix Relay Hold Pushbutton does not supply power to the TCB Shunt and UV Trip coils.

D. **Correct** - The Matrix Relay Hold Pushbutton supplies power to maintain the individual matrix relays (AB, AC, AD, etc. …) powered to prevent trip relays K-1 thru K-4 from opening (and opening their respective TCBs) during testing.

Question 40 (Q20201)			
Topic:	Function of the Matrix Relay Hold Pushbutton		
Tier/Group:	2/1		
K/A Info:	012 - Reactor Protection • K4 - Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: • K4.08 - Logic matrix testing		
RO Importance:	2.8*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	SD-058, Reactor Protective System		
Comments:	None		

41. 013 - Engineered Safety Features Actuation

U-1 was operating at 100% power when the 11 Main Feed Reg Valve goes shut concurrent with a loss of 11 120V Vital AC Bus (1Y01). 11 S/G water level is now (-)200 inches.

Which **ONE** of the following is the most complete list of components actuated?

A. Only MS-4070-CV opened

B. MS-4070-CV and MS-4071-CV opened

C. MS-4070-CV opened and 13 AFW pump started

D. MS-4070-CV and MS-4071-CV opened and 13 AFW pump started

Answer: A

Answer Explanation:

A. **Correct** - The loss of 11 120V Vital AC Bus results in the loss of power to the AFAS "A" Actuation Logic Cabinet and the AFAS ZD Sensor Cabinet; AFAS "A" will not be actuated. AFAS "B" Actuation Logic Cabinet remains powered and will actuate AFAS "B" at a S/G level of (-) 170 inches on 2/4 AFAS channels for at least 20 seconds. **Only** components receiving an AFAS "B" signal (1-MS-4070-CV) will actuate.

B. **Incorrect** - The loss of 11 120V Vital AC Bus results in the loss of power to the AFAS "A" Actuation Logic Cabinet and the AFAS ZD Sensor Cabinet; AFAS "A" will not be actuated. AFAS "B" Actuation Logic Cabinet remains powered and will actuate AFAS "B" at a S/G level of (-) 170 inches on 2/4 AFAS channels for at least 20 seconds. **Only** components receiving an AFAS "B" signal (1-MS-4070-CV) will actuate. 1-MS-4071-CV receives its actuation signal from the AFAS "A" logic cabinet.

C. **Incorrect** - The loss of 11 120V Vital AC Bus results in the loss of power to the AFAS "A" Actuation Logic Cabinet and the AFAS ZD Sensor Cabinet; AFAS "A" will not be actuated. AFAS "B" Actuation Logic Cabinet remains powered and will actuate AFAS "B" at a S/G level of (-) 170 inches on 2/4 AFAS channels for at least 20 seconds. **Only** components receiving an AFAS "B" signal (1-MS-4070-CV) will actuate. 13 AFW Pp receives its actuation signal from the AFAS "A" logic cabinet.

D. **Incorrect** - The loss of 11 120V Vital AC Bus results in the loss of power to the AFAS "A" Actuation Logic Cabinet and the AFAS ZD Sensor Cabinet; AFAS "A" will not be actuated. AFAS "B" Actuation Logic Cabinet remains powered and will actuate AFAS "B" at a S/G level of (-) 170 inches on 2/4 AFAS channels for at least 20 seconds. **Only** components receiving an AFAS "B" signal (1-MS-4070-CV) will actuate. 13 AFW Pp and 1-MS-4071-CV receive their actuation signals from the AFAS "A" logic cabinet.

Question 41 (Q97153)			
Topic:	ESFAS loss or malfunction on sensors detectors		
Tier/Group:	2/1		
K/A Info:	013 - Engineered Safety Features Actuation • K6 - Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: • K6.01 - Sensors and detectors		
RO Importance:	2.7*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	SD-036, Auxiliary Feedwater 1C04-ALM, Aux Feedwater and Computer Alarm Manual		
Comments:	None		

42. 022 - Containment Cooling

Which **ONE** of the following pressure setpoints would you expect Containment Air Coolers to automatically start?

(1) Containment Pressure
(2) RCS Pressure

A. (1) 2.8 PSIG;
(2) 1725 PSIA

B. (1) 2.8 PSIG;
(2) 1740 PSIA

C. (1) 4.25 PSIG;
(2) 1725 PSIA

D. (1) 4.75 PSIG;
(2) 1740 PSIA

Answer: B

Answer Explanation:

A. **Incorrect** - Containment Cooler Fans are started by SIAS. SIAS is actuated by a high Containment Pressure of 2.8 PSIG or a low RCS pressure of **1740 PSIA**.

B. **Correct** - Containment Cooler Fans are started by SIAS. SIAS is actuated by a high Containment Pressure of 2.8 PSIG or a low RCS pressure of 1740 PSIA.

C. **Incorrect** - Containment Cooler Fans are started by SIAS. SIAS is actuated by a high Containment Pressure of **2.8 PSIG** or a low RCS pressure of **1740 PSIA**. 4.25 PSIG is the setpoint for CSAS actuation which starts the Containment Spray Pumps. 1725 PSIA is the minimum RCS pressure SIAS must be actuated by per the Tech Specs.

D. **Incorrect** - Containment Cooler Fans are started by SIAS. SIAS is actuated by a high Containment Pressure of **2.8 PSIG** or a low RCS pressure of **1740 PSIA**. 4.75 PSIG is the maximum Containment pressure SIAS must be actuated by per the Tech Specs. 1725 PSIA is the minimum RCS pressure SIAS must be actuated by per the Tech Specs.

Question 42 (Q97169)			
Topic:	Match component actuation to ESFAS signals		
Tier/Group:	2/1		
K/A Info:	022 - Containment Cooling • A3 - Ability to monitor automatic operation of the CCS, including: • A3.01 - Initiation of safeguards mode of operation		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	1C08-ALM		
Comments:	None		

43. 022 - Containment Cooling

What are the **Technical Specification maximum** Containment Pressure setpoint limits for the actuation signals that start the following components?

(1) Containment Spray Pumps start
(2) Containment Spray CV's open

A. (1) ≤ 2.8 PSIG;
(2) ≤ 4.25 PSIG

B. (1) ≤ 2.8 PSIG;
(2) ≤ 4.75 PSIG

C. (1) ≤ 4.25 PSIG;
(2) ≤ 4.25 PSIG

D. (1) ≤ 4.75 PSIG;
(2) ≤ 4.75 PSIG

Answer: D

Answer Explanation:

A. **Incorrect** - 2.8 & 4.25 PSIG are the actual setpoints for SIAS (starts the Containment Spray Pumps) and CSAS (opens the Containment Spray CVs). The question asks for the Tech Spec limits for actuation signals which are both a Containment pressure of 4.75 PSIG.

B. **Incorrect** - **2.8 PSIG** is the actual setpoint for SIAS (starts the Containment Spray Pumps). 4.75 PSIG is the Tech Spec limit for CSAS (opens the Containment Spray CVs). The question asks for the Tech Spec limits for actuation signals which are both a Containment pressure of 4.75 PSIG.

C. **Incorrect** - **4.25 PSIG** is the actual setpoint for CSAS which opens the Containment Spray CVs. Per T.S. 3.3.4, Table 3.3.4-1, the SIAS Tech Spec limit is ≤ 4.75 PSIG and the CSAS Tech Spec limit is ≤ 4.75 PSIG

D. **Correct** - The Containment Spray Pumps are started by SIAS and the Containment Spray Header CVs are opened by CSAS. Per T.S. 3.3.4, Table 3.3.4-1, the SIAS Tech Spec limit is ≤ 4.75 PSIG and the CSAS Tech Spec limit is ≤ 4.75 PSIG

Question 43 (Q97154)			
Topic:	Containment Cooling auto operation on containment pressure		
Tier/Group:	2/1		
K/A Info:	022 - Containment Cooling • A1 - Ability to monitor automatic operation of the CCS, including: • A1.02 - Containment pressure		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - modified question		
Technical references:	T.S. 3.3.4, Table 3.3.4-1		
Comments:	Modified version of Q20788		

44. 026 - Containment Spray

During power operation a LOCA occurred. SIAS, CSAS and RAS have actuated.

Which **ONE** of the following functions are the Shutdown Cooling Heat Exchangers performing at this time?

A. Providing cooling of Containment Spray.

B. Providing a Shutdown Cooling flowpath.

C. Providing High Pressure Safety Injection cooling.

D. Providing Low Pressure Safety Injection cooling.

Answer: A

Answer Explanation:

A. **Correct** - RAS realigns CCW flow to the SDC Heat Exchanger to cool Containment Spray Pump discharge flow.

B. **Incorrect** - Use of the SDC Heat Exchangers as part of the Shutdown Cooling flowpath requires manual re-alignment of the LPSI Pumps.

C. **Incorrect** - HPSI flow does not pass through the SDC Heat Exchangers.

D. **Incorrect** - LPSI Pumps are secured automatically when RAS actuates. Use of SDC Heat Exchangers in support of LPSI cooling requires manual re-alignment of the LPSI Pump flowpaths and starting the LPSI Pumps.

Question 44 (Q17833)			
Topic:	SDC Heat Exchanger purpose during a LOCA		
Tier/Group:	2/1		
K/A Info:	026 - Containment Spray • K1 - Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: • K1.02 - Cooling water		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(8)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	Used in NRC exam administered 12/2008		
Technical references:	SD-052 & 061, Safety Injection and Containment Spray Systems		
Comments:	None		

45. 039 - Main and Reheat Steam

Given the following:

- A S/G Tube Rupture and Loss of Offsite Power occurred
- EOP-0 has been completed and the appropriate Optimal Recovery Procedure has been implemented.

Which **ONE** of the following describes the preferred method for cooldown of the RCS and when can the affected S/G be isolated?

A. TBV's;
T_{COLD} <515°F

B. ADV's;
T_{COLD} <515°F

C. TBV's;
T_{HOT} <515°F

D. ADV's;
T_{HOT} <515°F

Answer: D

Answer Explanation:

A. **Incorrect** - Because of the Loss of Offsite Power, the MSIVs will have been shut during the performance of EOP-0 making the **TBVs unavailable** to cooldown the RCS. Additionally, EOP-6 requires the RCS be cooled down to a $\mathbf{T_{HOT}}$ of <515°F. This action reduces the risk of challenging the steam generator safety valves of the affected S/G after it is isolated.

B. **Incorrect** - Because of the Loss of Offsite Power, the MSIVs will have been shut during the performance of EOP-0 necessitating the use of the ADVs to cooldown the RCS. Additionally, EOP-6 requires the RCS be cooled down to a $\mathbf{T_{HOT}}$ of <515°F. This action reduces the risk of challenging the steam generator safety valves of the affected S/G after it is isolated.

C. **Incorrect** - Because of the Loss of Offsite Power, the MSIVs will have been shut during the performance of EOP-0 making the **TBVs unavailable** to cooldown the RCS. Additionally, EOP-6 requires the RCS be cooled down to a T_{HOT} of <515°F. This action reduces the risk of challenging the steam generator safety valves of the affected S/G after it is isolated.

D. **Correct** - Because of the Loss of Offsite Power, the MSIVs will have been shut during the performance of EOP-0 making the TBVs unavailable to cooldown the RCS. Additionally, EOP-6 requires the RCS be cooled down to a T_{HOT} of <515°F. This action reduces the risk of challenging the steam generator safety valves of the affected S/G after it is isolated.

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Question 45 (Q97147)			
Topic:	Cooldown during a SGTR		
Tier/Group:	2/1		
K/A Info:	039 - Main and Reheat Steam • K1 - Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: • K1.02- Atmospheric relief dump valves		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EOP-6, Steam Generator Tube Rupture		
Comments:	None		

46. 059 - Main Feedwater

Unit-2 is escalating in power, recovering from a mid-cycle forced outage. The reactor is at approximately 50% power with 21 SGFP in operation. 22 SGFP is out of service for maintenance.

Under these conditions, which **ONE** of the following sets of parameters on 21 SGFP would support a decision to continue operation at current power level?

A. Suction flow: 15,200 GPM;
Turbine speed: 5060 RPM;
Suction pressure: 242 PSIG

B. Suction flow: 16,200 GPM;
Turbine speed: 5080 RPM;
Suction pressure: 262 PSIG

C. Suction flow: 17,200 GPM;
Turbine speed: 5335 RPM;
Suction pressure: 272 PSIG

D. Suction flow: 18,200 GPM;
Turbine speed: 5260 RPM;
Suction pressure: 252 PSIG

Answer: B

Answer Explanation:

Per the Precautions stated in OI-12A-2 one pump operation is permitted if all of the following conditions are maintained:

- Suction flow rate below 18,000 GPM
- Turbine speed below 5100 RPM
- Suction pressure above 250 PSIG

A. **Incorrect** - Suction pressure is **below** the stated operating limit of 250 PSIG

B. **Correct** - Per the Precautions stated in OI-12A-2

C. **Incorrect** - SGFP speed is **above** the stated operating limit of 5100 RPM
Plausible because speed is within the operating range for a **Unit-1** SGFP

D. **Incorrect** - SGFP speed is **above** the stated operating limit of 5100 RPM and SGFP **suction flow rate** is also above the limit for single pump operation
Plausible because speed is within the operating range for a **Unit-1** SGFP

Question 46 (Q97170)			
Topic:	Operating limitations on U-2 SGFPs		
Tier/Group:	2/1		
K/A Info:	059 - Main Feedwater • A1 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: • A1.03 – Power level restrictions for operation of MFW pumps and valves		
RO Importance:	2.7*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OI-12A-1, Feedwater System OI-12A-2, Feedwater System		
Comments:	None		

47. 061 - Auxiliary/Emergency Feedwater

Which **ONE** of the following describes the cooling/lubrication methods for (1) the Steam Driven AFW Pump bearing and (2) the AFW Pump Turbine bearings and (3) Conditions needed to auto start AFW pumps?

A. (1) Air cooled rotating slinger ring;
(2) Air cooled rotating slinger ring
(3) -170 inches for ≥20 seconds on 2 of 4 channels on either S/G

B. (1) Air cooled rotating slinger ring;
(2) AFW cooled rotating slinger ring supplemented by forced oil flow
(3) -170 inches for ≥20 seconds on 2 of 4 channels on either S/G

C. (1) AFW cooled rotating slinger ring supplemented by forced oil flow;
(2) Air cooled rotating slinger ring
(3) -170 inches for ≥20 seconds on 2 of 4 channels on both S/G's

D. (1) AFW cooled rotating slinger ring supplemented by forced oil flow;
(2) AFW cooled rotating slinger ring supplemented by forced oil flow
(3) -170 inches for ≥20 seconds on 2 of 4 channels on both S/G's

Answer: B

Answer Explanation:

A. **Incorrect** - Per SD-036: the SDAFW Pp bearings are cooled and lubricated by an air cooled rotating slinger ring and the SDAFW **Turbine Bearings are cooled by AFW flow and lubricated by a rotating slinger ring supplemented by forced oil flow from a saddle mounted pump.** . -170" 2 of 4 channels on either S/G generates AFAS which starts AFW Pumps

B. **Correct** - Per SD-036: the SDAFW Pp bearings are cooled and lubricated by an air cooled rotating slinger ring and the SDAFW Turbine Bearings are cooled by AFW flow and lubricated by a rotating slinger ring supplemented by forced oil flow from a saddle mounted pump. -170" 2 of 4 channels on either S/G generates AFAS which starts AFW Pumps

C. **Incorrect - Per SD-036: the SDAFW Pp bearings are cooled and lubricated by an air cooled rotating slinger ring and the SDAFW Turbine Bearings are cooled by AFW flow and lubricated by a rotating slinger ring supplemented by forced oil flow from a saddle mounted pump. -170" 2 of 4 channels on either S/G generates AFAS which starts AFW Pumps**

D. **Incorrect** - Per SD-036: **the SDAFW Pp bearings are cooled and lubricated by an air cooled rotating slinger ring** and the SDAFW Turbine Bearings are cooled by AFW flow and lubricated by a rotating slinger ring supplemented by forced oil flow from a saddle mounted pump. **-170" 2 of 4 channels on either S/G generates AFAS which starts AFW Pumps.**

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Question 47 (Q97148)			
Topic:	AFW design feature cooing and lubrication		
Tier/Group:	2/1		
K/A Info:	061 - Auxiliary/Emergency Feedwater • K4 - Knowledge of AFW design feature(s) and/or interlock(s) which provide for the following: • K4.13 - Initiation of cooling water and lube oil		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	SD-036, Auxiliary Feedwater		
Comments:	None		

48. 062 - AC Electrical Distribution

U-1 is in a normal electrical lineup with exception of 1Y02 has been shifted to the Inverter Back-up bus, 1Y11. Subsequently, a loss of the Red Bus occurs.

Which **ONE** of the following statements describe the effect on plant equipment?

A. 11 4kV bus remains energized from P-13000-1;
1Y11 remains powered from MCC-114;
1Y02 remains energized.

B. 1Y11 is de-energized due to the loss of MCC-104;
1B DG starts and automatically repowers 14 4kV Bus;
1Y02 is re-energized.

C. 1Y11 is de-energized due to the loss of MCC-114;
1A DG starts and automatically repowers 11 4kV Bus;
1Y02 is re-energized.

D. 1Y11 is de-energized due to the loss of P-13000-2;
1B DG must be manually started and loaded;
1Y02 is re-energized.

Answer: D

Answer Explanation:

A. **Incorrect** - 1Y11 (inverter backup bus) is powered from MCC 104, normally supplied by 14 4kV bus through P-13000-2 via the red bus.

B. **Incorrect** - May be selected if it is not recognized that the ESFAS B logic cabinet has been deenergized.

C. **Incorrect** - Could be selected if there is confusion on components supplied by Red Bus/Black bus. DGs do not automatically repower 4KV Busses when the Inverter B/U Bus is in service and lost.

D. **Correct** - 1Y11 (inverter backup bus) is powered from MCC 104, normally supplied by 14 4kV bus through P-13000-2 via the red bus. The 1B DG would not start because the ESFAS "B" logic cabinet is deenergized (results in no U/V start signal actuation). Manually starting the 1B DG is needed to reenergize MCC-104 which reenergizes 1Y11 and restores power to 1Y02.

Question 48 (Q92672)			
Topic:	Loss of red bus when 1Y02 is aligned to inverter backup bus		
Tier/Group:	2/1		
K/A Info:	062 - AC Electrical Distribution • K3 - Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: • K3.02 - ED/G		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:			
Comments:	None		

49. 063 - DC Electrical Distribution

Given the following conditions:

- BOTH Units are operating at 100% power
- 1C33 Alarm Window, T-20, "125V DC Ground Detected" is illuminated

Which **ONE** of the following describes the DC Busses potentially affected by this alarm, and the indication observed on the bus that has a ground?

A. DC Busses 11, 12, 21, and 22;
Red LED on Ground Detection Panel will be ILLUMINATED.

B. DC Busses 11, 12, 21, and 22;
Red LED on Ground Detection Panel will be EXTINGUISHED.

C. DC Busses 11 and 21 ONLY;
Red LED on Ground Detection Panel will be EXTINGUISHED.

D. DC Busses 11 and 21 ONLY;
Red LED on Ground Detection Panel will be ILLUMINATED.

Answer: D

Answer Explanation:

A. **Incorrect** - **Only 125VDC buses 11 and 21 are monitored by Annunciator window "T-20"** and Red LED on Ground Detection Panel will be ILLUMINATED when a ground is present.

B. **Incorrect** - **Only 125VDC buses 11 and 21 are monitored by Annunciator window "T-20"** and **Red LED on Ground Detection Panel will be ILLUMINATED when a ground is present.**

C. **Incorrect** - Only 125VDC buses 11 and 21 are monitored by Annunciator window "T-20" and **Red LED on Ground Detection Panel will be ILLUMINATED when a ground is present.**

D. **Correct** - Only 125VDC buses 11 and 21 are monitored by Annunciator window "T-20" and Red LED on Ground Detection Panel will be ILLUMINATED when a ground is present.

Question 49 (Q74605)			
Topic:	Monitor and operate the 125 VDC, 120 VAC, and instrument AC Systems		
Tier/Group:	2/1		
K/A Info:	063 - DC Electrical Distribution • A3 - Ability to monitor automatic operation of the DC electrical system, including: • A3.01 - Meters, annunciators, dials, recorders, and indicating lights		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	1C33 Alarm Response Manual		
Comments:	None		

50. 064 - Emergency Diesel Generator

Safety Related Diesel Generator auxiliaries get power from which of the following buses?

Diesel Generator	**Bus**
1A ____	1) MCC-104
1B ____	2) MCC-114
2A ____	3) MCC-122
2B ____	4) MCC-123
	5) MCC-204
	6) MCC-214

A. 1, 2, 5, 6
B. 2, 1, 6, 5
C. 4, 1, 6, 5
D. 3, 2, 5, 6

Answer: C

Answer Explanation:

A. **Incorrect - 1A D/G Auxiliaries are powered from MCC-123,**
1B D/G Auxiliaries are powered from MCC 1BG through MCC-104
2A D/G Auxiliaries are powered from MCC 2AG through MCC-214
2B D/G Auxiliaries are powered from MCC 2BG through MCC-204

B. **Incorrect - 1A D/G Auxiliaries are powered from MCC-123,**
1B D/G Auxiliaries are powered from MCC 1BG through MCC-104
2A D/G Auxiliaries are powered from MCC 2AG through MCC-214
2B D/G Auxiliaries are powered from MCC 2BG through MCC-204

C. **Correct** - 1A D/G Auxiliaries are powered from MCC-123,
1B D/G Auxiliaries are powered from MCC 1BG through MCC-104
2A D/G Auxiliaries are powered from MCC 2AG through MCC-214
2B D/G Auxiliaries are powered from MCC 2BG through MCC-204

D. **Incorrect - 1A D/G Auxiliaries are powered from MCC-123,**
1B D/G Auxiliaries are powered from MCC 1BG through MCC-104
2A D/G Auxiliaries are powered from MCC 2AG through MCC-214
2B D/G Auxiliaries are powered from MCC 2BG through MCC-204

Question 50 (Q97149)			
Topic:	Knowledge of D/G auxiliary power supplies		
Tier/Group:	2/1		
K/A Info:	064 - Emergency Diesel Generator • K2 - Knowledge of bus power supplies to the following: • K2.02 - Fuel oil pumps		
RO Importance:	2.8*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OI-27D-2, Station Power 480 Volt System Breaker Lineup		
Comments:	None		

51. 073 - Process Radiation Monitoring

While discharging 11 WGDT to U-2, which **ONE** of the following conditions would **allow** the discharge to continue?

A. Critical high computer alarm 2R5415A! (U-2 Main Vent RMS)

B. Alert high computer alarm 0-R2191! (Waste Gas RMS)

C. 12 WGDT low pressure computer alarm actuates

D. Running U-2 Main Exhaust fan trips

Answer: B

Answer Explanation:

A. **Incorrect** - OI-17B Waste Gas Sect. 6.4 "Waste Gas Release "specifically requires securing release immediately if an RMS critical setpoint is exceeded.

B. **Correct** - OI-17B Waste Gas Sect. 6.4 "Waste Gas Release" directs entering setpoints IAW Waste Gas discharge permit. Alert Alarm is just a warning and discharge can continue.

C. **Incorrect** - OI-17B Waste Gas Sect. 6.4 "Waste Gas Release" directs setting low pressure alarm for WGDT's not being discharged to 5 psig below current pressure. If 12 WGDT low pressure alarm comes in then the wrong WGDT may be discharging and release must be secured.

D. **Incorrect** - Waste Gas release permit directs Main Vent Exhaust fan running, if the fan tripped off there is no auto start of other fan so discharge must be secured.

Question 51 (Q97150)			
Topic:	RMS operate/monitor from control room during effluent release		
Tier/Group:	2/1		
K/A Info:	073 - Process Radiation Monitoring • A4 - Ability to manually operate and/or monitor in the control room: • A4.01 - Effluent release		
RO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OI-17B Waste Gas System		
Comments:	Student must analyze 4 different conditions and determine if the release can continue. Therefore this question is considered upper cognitive level		

52. 073 - Process Radiation Monitoring

U-1 commenced a fuel shuffle 2 hours ago with containment purge in service when an RMS alarm occurs. Investigation shows 1-RI-5316C (Containment Area) is saturated-pegged high.

Which **ONE** of the following describes the impact on the plant and the actions required to complete the fuel shuffle?

A. Containment Purge automatically isolates;
Pull fuses for RMS then reset RMS after inserting fuse

B. Containment Purge automatically isolates;
Place detector in Level Cal and continue to move fuel with the 3 remaining detectors

C. Containment Purge continues to operate;
Pull fuses for RMS then reset RMS after inserting fuse

D. Containment Purge continues to operate;
Place detector in Level Cal and continue to move fuel with the 3 remaining detectors

Answer: C

Answer Explanation:

A. **Incorrect** - Containment Radiation Signal (CRS) is actuated by 2 of 4 Containment area rad monitors, which would isolate Containment Purge. **Only one rad monitor fails high therefore no securing of Containment Purge**. OI-35 Radiation Monitoring System Sect 6.2 "Normal Operation" directs removing fuse for saturated detector and resetting detector after inserting fuse. No need to secure fuel handling based on a single pegged high area rad monitor.

B. **Incorrect** - Containment Radiation Signal (CRS) is actuated by 2 of 4 Containment area rad monitors, which would isolate Containment Purge. **Only one rad monitor fails high therefore no securing of Containment Purge**. OI-35 Radiation Monitoring System Sect 6.2 "Normal Operation" directs removing fuse for saturated detector and resetting detector after inserting fuse. **Moving fuel with purge on with 3 detectors is not per the procedure. Need all 4 .**

C. **Correct** - Containment Radiation Signal (CRS) is actuated by 2 of 4 Containment area rad monitors, which would isolate Containment Purge. Only one rad monitor fails high therefore no securing of Containment Purge. OI-35 Radiation Monitoring System Sect 6.2 "Normal Operation" directs removing fuse for saturated detector and resetting detector after inserting fuse. No need to secure fuel handling based on a single pegged high area rad monitor.

D. **Incorrect** - Containment Radiation Signal (CRS) is actuated by 2 of 4 Containment area rad monitors, which would isolate Containment Purge. Only one rad monitor fails high therefore no securing of Containment Purge. OI-35 Radiation Monitoring System Sect 6.2 "Normal Operation" directs removing fuse for saturated detector and resetting detector after inserting fuse. **Moving fuel with purge on with 3 detectors is not per the procedure. Need all 4**

Question 52 (Q97151)			
Topic:	RMS predict impact then mitigate detector failure		
Tier/Group:	2/1		
K/A Info:	073 - Process Radiation Monitoring • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.02 - Detector failure		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OI-35 Radiation Monitoring System SD-48 ESFAS System Description		
Comments:	None		

53. Components affected by loss of 12 SRW header

Which **ONE** of the following components will be affected by the loss of 12 SRW header?

A. EHC oil coolers

B. 1B Diesel Generator

C. 12 Containment Air Cooler

D. 12 Blowdown Heat Exchanger

Answer: B

Answer Explanation:

A. **Incorrect** - EHC cooler supplied from 11 SRW Header

B. **Correct** - 1B Diesel Generator is supplied by 12 SRW Header

C. **Incorrect** - 12 Containment Air Cooler supplied from 11 SRW header

D. **Incorrect** - 12 Blowdown Heat Exchanger supplied from 11 SRW header

Question 53 (Q74576)			
Topic:	Components affected by loss of 12 SRW header		
Tier/Group:	2/1		
K/A Info:	076 Service Water • K1 - Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems: • K1.08 - RHR system		
RO Importance:	3.5*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(8)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-7B-1		
Comments:	SRW system does not directly interface with RHR at CCNPP, but SFP cooling is can be credited as a decay heat removal system by aligning the SFPHX to cool the RFP. This question tests student's knowledge of physical connection between SRW and SFP Cooling.		

54. 078 - Instrument Air

What design features allow the 11 Salt Water Air Compressor to be operated during a Control Room fire?

A. Key operated Local/Remote handswitch located at MCC-104
Local Control hand switch located at the Safe Shutdown Panel

B. Key operated Local/Remote handswitch located at MCC-114,
Local Control hand switch located at the Safe Shutdown Panel

C. Key operated Local/Remote handswitch located at MCC-104,
Local Control handswitch located at MCC-104

D. Key operated Local/Remote handswitch located at MCC-114,
Local Control handswitch located at MCC-114

Answer: D

Answer Explanation:

A. **Incorrect – 11 SWAC Local/Remote keyswitch and local control handswitch are both located on MCC-114**

B. **Incorrect** - 11 SWAC Local/Remote keyswitch and **local control handswitch** are both located on MCC-114

C. **Incorrect - 11 SWAC Local/Remote keyswitch and local control handswitch are both located on MCC-114**

D. **Correct** - 11 SWAC Local/Remote keyswitch and local control handswitch are both located on MCC-114

Question 54 (Q20332)			
Topic:	Operate the SWACs locally		
Tier/Group:	2/1		
K/A Info:	078 - Instrument Air • K2 - Knowledge bus power supplies to the following: • K2.02 - Emergency Air compressor		
RO Importance:	3.3*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-9A-1		
Comments:	None		

55. 103 - Containment

Which **ONE** of the following conditions, by itself, is a loss of Containment Integrity?

A. 2-SRW-1582-CV, 21 CAC EMER DISCH, is open

B. 2-PS-5464-CV, RCS SAMPLE ISOL, is open

C. 2-EAD-5463-MOV, CNTMT NORM SUMP DRN, is open

D. 2-SI-4144-MOV, CNTMT SUMP OUTLET ISOL, is open

Answer: D

Answer Explanation:

A. **Incorrect** - Containment Cooler SRW isolation valves get ESFAS signals. As long as the valves are operable, containment integrity is maintained.

B. **Incorrect** - , RCS Sample Isolation gets an ESFAS signal. As long as the valve is operable, containment integrity is maintained.

C. **Incorrect** - The normal containment sump valves get an ESFAS signal which will automatically shut the valves, so containment integrity is maintained.

D. **Correct** - Per STP-O-55-2, Att. 2, 2-SI-4144 must be shut.

Question 55 (Q74932)			
Topic:	Loss of Containment Integrity		
Tier/Group:	2/1		
K/A Info:	103 - Containment • K3 - Knowledge of the effect that a loss or malfunction of the containment system will have on the following: • K3.02 - Loss of containment integrity under normal operations		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	Used in NRC exam administered 12/2008		
Technical references:	STP-O-55-2, Containment Integrity Verification, Mode 1 - 4		
Comments:	None		

56. 001 - Control Rod Drive

Which **ONE** of the following set of conditions results in a CEA Withdrawal Prohibit (CWP) signal?

A. 2 out of 4 pre-trips actuated from VOPT, High SUR or TM/LP condition.

B. 2 out of 4 pre-trips actuated from APD, TM/LP or VOPT condition.

C. 1 out of 4 pre-trips actuated from TM/LP, APD or High SUR condition.

D. 1 out of 4 pre-trips actuated from High SUR, VOPT or APD condition.

Answer: A

Answer Explanation:

A. **Correct** - Per Alarm Response Manual window D-35

B. **Incorrect** - Logic is **2** out of 4 pre-trips actuated from VOPT, **High SUR** or **TM/LP** condition.

C. **Incorrect** - Logic is **2** out of 4 pre-trips actuated from **VOPT**, High SUR or TM/LP condition.

D. **Incorrect** - Logic is **2** out of 4 pre-trips actuated from VOPT, High SUR or **TM/LP** condition.

Question 56 (Q97171)			
Topic:	Conditions causing a CWP		
Tier/Group:	2/2		
K/A Info:	001 - Control Rod Drive • K1 - Knowledge of the physical connections and/or cause-effect relationships between the CRDS and the following systems: • K1.05 - NIS and RPS		
RO Importance:	4.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	Alarm Response Manual 1C05		
Comments:	None		

57. 002 - RCS

Which **ONE** of the following would cause a loss of conservatism in relation to the RCS flow trip from RPS?

A. Grid frequency lowers to 59.9 Hz

B. Grid frequency rises to 60.1 Hz

C. Debris build up in primary side of S/G causes 5 tubes to plug

D. Calibration of 1-PDI-101B (RCS Flow Indicator) is drifting low

Answer: C

Answer Explanation:

A. **Incorrect** – Changing actual flow does not affect the conservatism of the setpoint

B. **Incorrect** - Changing actual flow does not affect the conservatism of the setpoint

C. **Correct** - This condition results in indicated flow appearing higher than actual flow, which will cause flow to be lower than RPS sensed when a trip occurs

D. **Incorrect** - Indicated flow goes down with actual flow staying the same, low flow trip setpoint remains the same, therefore closer to low flow trip which is more conservative

Question 57 (Q97177)			
Topic:	RCS Flow indication effect on indicated / actual reactor power		
Tier/Group:	2/2		
K/A Info:	002 - RCS • A1 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCS controls including: • A1.05 - RCS Flow		
RO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	TS 3.3.1 Basis		
Comments:	None		

58. 011 - Pressurizer Level Control System

U-2 tripped 2 minutes ago due to a Loss of the Red Bus. No operator actions or additional failures have occurred. Which Charging Pump(s) if any are running?

A. 21, 22, 23 Charging Pumps

B. 21, 23 Charging Pumps

C. 22 Charging Pump

D. No Charging Pumps

Answer: B

Answer Explanation:

- 21 charging pump will be running because powered from black bus (not a D/G) and power lost to relay in PZR level control ckt which tells all charging pumps to start.
- 22 charging pump is not running because powered from 2B D/G. Charging pumps powered from a D/G require reset of charging spring to start (per question stem not done)
- 23 charging pump will be running because normally powered from 21 480V bus (same as 21 charging pump), power lost to relay in PZR level control ckt which tells all charging pumps to start.

A. **Incorrect** - 22 pump is not running, see explanation above

B. **Correct** - See explanation above

C. **Incorrect** - 22 pump is not running, see explanation above

D. **Incorrect** - 21 & 23 pumps running

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Question 58 (Q97173)			
Topic:	Loss of electrical effects on Charging Pumps		
Tier/Group:	2/2		
K/A Info:	011 - Pressurizer Level Control System • K6- Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS: • K6.04 - Operation of PZR level controllers		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power		
Comments:	None		

59. Knowledge of IRU power supplies

An electrical bus fault occurred which caused 21 Charging Pump to lose power.

Which **ONE** of the following pieces of equipment will also be de-energized?

A. 12 Control Room A/C Compressor

B. 21 Battery Charger

C. 23 Iodine Removal Unit

D. 23 Containment Air Cooler

Answer: C

Answer Explanation:

A. **Incorrect** - 21 Charging Pump is powered from 480V Load Center 21B; 12 Control Room A/C Compressor is powered from **480V Load Center 24A**

B. **Incorrect** - 21 Charging Pump is powered from 480V Load Center 21B; 21 Battery Charger is powered from **480V Load Center 24A**

C. **Correct** - 21 Charging Pump and 23 Containment Filter (IRU) are powered from 480V Load Center 21B

D. **Incorrect** - 21 Charging Pump is powered from 480V Load Center 21B; 23 Containment Air Cooler is powered from **480V Load Center 24A**

Question 59 (Q91758)			
Topic:	Knowledge of IRU power supplies		
Tier/Group:	2/2		
K/A Info:	027 - Containment Iodine Removal • K2 - Knowledge of bus power supplies to the following: • K2.01 - Fans		
RO Importance:	3.1*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	OI-27D-2, Station Power 480 Volt System Breaker Lineup		
Comments:	None		

60. 028 - Hydrogen Recombiner and Purge Control

Following a Loss of Coolant Accident, the ________(1)__________ are placed in service at a Hydrogen concentration in the Containment of __(2)__% to **Prevent Exceeding** the lower flammability limit.

A. (1) Hydrogen Recombiners
(2) 0.5

B. (1) Hydrogen Recombiners and Hydrogen Purge
(2) 0.5

C. (1) Hydrogen Recombiners;
(2) 4.0

D. (1) Hydrogen Recombiners and Hydrogen Purge;
(2) 4.0

Answer: A

Answer Explanation:

EOP-5 directs use of the H_2 Recombiners if H_2 concentrations rises to **.5%** and goes on to state if H_2 concentration rises to 4% consideration should be given to securing the H_2 Recombiners and placing the H_2 Purge System in operation (based on Technical Support Center recommendation). Design of both systems is that either will maintain H_2 concentration below the flammability limit of 4%.

A. **Correct** - H_2 Recombiners are placed in service if Containment H_2 concentration reaches .5%. Design of both systems is that either will maintain H_2 concentration below the flammability limit of 4%.

B. **Incorrect** - H_2 Recombiners are **placed in service** if Containment H_2 concentration reaches .5%. Design of both systems is that **either** will maintain H_2 concentration below the flammability limit of 4%.

C. **Incorrect** - H_2 Recombiners are placed in service if Containment H_2 concentration reaches **.5%**. Design of both systems is that either will maintain H_2 concentration below the flammability limit of 4%.

D. **Incorrect** - H_2 Recombiners are **placed in service** if Containment H_2 concentration reaches **.5%**. Design of both systems is that **either** will maintain H_2 concentration below the flammability limit of 4%.

Question 60 (Q97174)			
Topic:	Post-accident Containment H_2 control		
Tier/Group:	2/2		
K/A Info:	028 - Hydrogen Recombiner and Purge Control • K5 - Knowledge of the operational implications of the following concepts as they apply to the HRPS: • K5.02 - Flammable hydrogen concentration		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	EOP-5, Loss of Coolant Accident EOP-5, Loss of Coolant Accident Technical Basis document SD-060C, Hydrogen Purge System Description		
Comments:	None		

61. 029 - Containment Purge

Given the following:

- Unit-1 is in Mode 6
- A Containment Normal Purge is in progress
- Refueling operations are in progress
- A fuel assembly is dropped within the core
- Containment Area Radiation Monitors (1-RE-5316-A through D) are in alarm at 200 mrem/hr.

What automatic actions occur?

A. Containment Purge Supply and Exhaust CVs SHUT;
All Iodine Removal Units receive a START signal;
Hydrogen Purge Outlet Containment isolation MOVs SHUT.

B. Containment Purge Supply and Exhaust CVs SHUT;
Containment Purge Supply and Exhaust Fans STOP;
Hydrogen Purge Outlet Containment isolation MOVs SHUT.

C. Containment Purge Exhaust CVs SHUT;
Penetration Room Exhaust Fans receive START signal;
Hydrogen Purge Outlet Containment Isolation MOVs SHUT

D. Containment Purge Supply and Exhaust Fans STOP;
Penetration Room Exhaust Fans receive a START signal;
All Iodine Removal Units receive a START signal.

Answer: B

Answer Explanation:

A. **Incorrect** - The IRUs do not receive an automatic signal from the CRS circuitry.

B. **Correct** - A CRS is initiated by 2/4 coincidence from RI-5316A thru D. These six components actuate to the specified positions per LD-58A

C. **Incorrect** - The Purge Supply CV shuts and the Penetration Room Exhaust Fans do not receive an automatic signal from the CRS circuitry.

D. **Incorrect** - The Purge Supply and Exhaust CVs shut and the Penetration Room Exhaust Fans and IRUs do not receive an automatic signal from the CRS circuitry.

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Question 61 (Q24719)			
Topic:	CRS Actuation		
Tier/Group:	2/2		
K/A Info:	029 - Containment Purge • A3 - Ability to monitor automatic operation of the Containment Purge System including: • A3.01 - CPS isolation		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	Used in NRC exam administered 12/2008		
Technical references:	SD-048, Engineered Safety Features Actuation System Description SD-060B, Containment Purge System Description		
Comments:	None		

62. 035 - Steam Generator System

Unit 2 is operating at 70% power when a Feedwater System perturbation occurs.

Which **ONE** of the following sets of conditions would **require** a Reactor Trip?

A. “SGFP(S) SUCT PRESS LO” alarm annunciates

B. “SGFPT TRIP” alarm annunciates and 22 SGFP cannot be reset

C. "SG LVL LO CH PRE-TRIP" alarm annunciates; feed flow is greater than steam flow

D. "21 SG FW CONTR CH LVL" alarm annunciates; S/G level is approaching (-)40 inches

Answer: D

Answer Explanation:

A. **Incorrect** - Receipt of the “SGFP(S) SUCT PRESS LO” alarm is not, by itself, sufficient criteria for tripping the unit. The alarm setpoint of 256 PSIG provides some margin to the trip setpoint which is 221 PSIG (2/2 logic) for greater than 20 seconds.

B. **Incorrect** - A reactor trip would not be ordered for a SGFT trip at the stated power level. One SGFPT is capable of providing required feedwater flow at the stated power.

C. **Incorrect** - "SG LVL LO CH PRE-TRIP" alarm annunciates; feed flow is greater than steam flow

D. **Correct** - AOP-3G, Malfunction Of Main Feedwater System, supplies the following trip criteria: **IF** SG level is approaching (-)40 inches, **THEN**, with the approval of the SM/CRS, trip the reactor and implement EOP-0, Post Trip Immediate Actions

Question 62 (Q97195)			
Topic:	S/G Level related trip criteria		
Tier/Group:	2/2		
K/A Info:	035 - Steam Generator System • 2.4 - Emergency Procedures / Plan • 2.4.45 - Ability to prioritize and interpret the significance of each annunciator alarm		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	1C03-ALM, Condensate & Feedwater Control Alarm Manual AOP-3G, Malfunction Of Main Feedwater System		
Comments:	None		

63. 041 - Steam Dump/Turbine Bypass Control

Unit 1 is operating at 100% power, with a Core Burnup of 1200 MWD/MTU, when an ADV fails open.

What is the effect on Unit 1 and what is/are the **first** action(s) taken?

A. Reactor power rises causing a Reactor trip on VOPT;
Implement EOP-0.

B. Reactor power lowers;
Raise turbine load to restore T_{COLD} to the program band.

C. Reactor power rises but reactor does NOT trip;
Dispatch operators to manually isolate the ADV and insert CEAs as necessary to control reactor power.

D. Reactor power rises but reactor does NOT trip;
Insert CEAs as necessary to control reactor power and reduce turbine load to restore T_{COLD} to the program band.

Answer: D

Answer Explanation:

A. **Incorrect** - ADV will raise power approximately three percent which is below the VOPT setpoint. Plausible because power does rise just not enough to cause trip

B. **Incorrect** - Raising turbine load would decrease T_{COLD} further and cause reactor power to raise further. Plausible because BOL has +MTC at lower power

C. **Incorrect** - Steps to dispatch operators to manually isolate the ADV follow the correct action of reducing turbine load to maintain T_{COLD} in the program band. Plausible because everything correct except isolating ADV comes after insert CEA and adjust turbine load

D. **Correct** - Per AOP-7K, Section IV. One ADV is good for 2.5% steam demand, so the VOPT trip setpoint should not be reached and no trip on high power should occur.

Question 63 (Q28429)			
Topic:	ADV fails open in Mode 1		
Tier/Group:	2/2		
K/A Info:	041 - Steam Dump/Turbine Bypass Control • K3 - Knowledge of the effect that a loss or malfunction of the SDS will have on the following: • K3.02 - RCS		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	Used in NRC exam administered 12/2008		
Technical references:	AOP-7K, Overcooling Event in Mode One or Two		
Comments:	None		

64. 045 - Main Turbine Generator

Why is "IMPULSE IN" selected when performing Governor Valve (GV) testing on U-2?

A. To complete the circuit that allows shutting governor valves individually

B. To bypass the Valve Position Limit circuit to the GV's not being tested.

C. To provide for more linear and faster response to changes in load during testing

D. To limit the opening of GV's during testing

Answer: C

Answer Explanation:

A. **Incorrect** - plausible as pushbutton to perform testing

B. **Incorrect** - plausible because VPL is manually raised to max setting for valve testing

C. **Correct** - Depressing the Imp In pushbutton enables impulse pressure feedback to the control circuit for better control

D. **Incorrect** - plausible because function of VPL (valve position limit)

Higher cognitive due to fact that students will not have this memorized and must analyze system operation and procedure requirements to ascertain the correct answer

Question 64 (Q97175)			
Topic:	Knowledge of "Impulse In" circuit		
Tier/Group:	2/2		
K/A Info:	045 - Main Turbine Generator • K4 - Knowledge of MT/G system design feature(s) and/or interlock (s) which provide for the following: • K4.01 - Programmed controller for relationship between steam pressure at T/G inlet (impulse, first stage) and plant power level		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OI-43C-2, Unit-2 Main Turbine Performance Evaluation Checks SD-93C-2 U-2 Main Turbine Controls		
Comments:	None		

65. 068 - Liquid Radwaste

Which **ONE** of the following conditions would **REQUIRE** a discharge of 12 RCWMT be secured?

A. Composite sampler fails

B. 0-RI-2201 RMS indication below background for >1 min

C. Level in 12 RCWMT reaches 1.5 feet

D. Discharge flow indicator fails

Answer: B

Answer Explanation:

A. **Incorrect** - Composite sampler failure is a problem that could result in securing discharge but not required.

B. **Correct** - Per OI-17C-4, IF the discharge activity indicated by the liquid waste RMS exhibits a sustained decrease below the discharge permit background values for greater than 30 seconds, THEN STOP the running RCWMT PP.

A. **Incorrect** - An RCWMT level of 1ft does not require securing the discharge. A Level of 0.9 feet will auto stop the selected RCWMT PP. Discharge of 12 RCWMT can continue to the auto-stop setpoint.

B. **Incorrect** - Discharge flow indicator is a problem that could result in securing the discharge but does not require securing the discharge

Question 65 (Q97176)			
Topic:	Discharge termination criteria		
Tier/Group:	2/2		
K/A Info:	068 - Liquid Radwaste • A4 - Ability to manually operate and/or monitor in the control room: • A4.02 - Remote radwaste release		
RO Importance:	3.2*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	OI-17C-4, Discharging Reactor Coolant Waste Tanks To The Environment		
Comments:	None		

66. 2.1 - Conduct of Operations

Unit-1 is at 100% power when 11 SW pump trips. "U-1 4KV MOTOR OVERLOAD" alarm is annunciated. No common mode failure is indicated and the Control Room crew has decided to align the standby pump per the applicable procedure.

Per plant administrative procedures, which **ONE** of the following describes the minimum acceptable direction given to the Plant Operator(s) to guide his response in this situation?

A. Direct the PPO to "Restore Saltwater per AOP-7A, Section C".

B. Direct the Outside Operator to "Align 13 Saltwater Pp to 11 SW Header per AOP-7A-1, Section V.C.2.B, on page 11".

C. Announce over the plant page "11 Saltwater Pp has tripped, realign 13 Saltwater Pp to 11 Saltwater Header".

D. Direct the Turbine Building Operator to "align 13 Saltwater Pp power supply to 11 4KV bus" and the outside operator to "align 13 Saltwater Pp to 11 header per AOP-7A-1"

Answer: B

Answer Explanation:

A. **Incorrect** - Provides no direction as to what components to restore. Unit, page number and specific steps of the procedure to use are not specified. This is not in compliance with the requirements of NO-1-200

B. **Correct** - Per NO-1-200: When directing the performance of procedural steps to Operators or other plant personnel who have a copy of the procedure, the specific steps should be identified as a minimum by affected Unit, procedure number, page number, and step or section number.

C. **Incorrect** - Per NO-1-200: Except during emergency conditions requiring immediate PO actions, page announcements shall not be used for passing operating orders. If orders are given via the plant page, in an emergency condition, the requirements of NO-1-200, step 5.32.A. are still applicable.

D. **Incorrect** - AOP-7A-1 will not direct shifting power supplies if power is available to the swing pump. No section #, page # or step #s are provided

Question 66 (Q20575)			
Topic:	Providing direction to personnel outside the CR		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.1 - Conduct of Operations • 2.1.8 - Ability to coordinate personnel activities outside the control room.		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	Used in NRC exam administered 7/2002		
Technical references:	NO-1-200, Control of Shift Activities AOP-7A-2, Loss Of Saltwater Cooling		
Comments:	None		

67. 2.1 - Conduct of Operations

Core offload is in progress. A core to upender fuel move sequence is occurring. During withdrawal of a fuel assembly from the core, as the hoist box position encoder readout is approaching 126.00 inches, the load increases from 1385 pounds to 2680 pounds.

Which **ONE** of the following explains why the load readout has increased?

A. The hoist box load is transferred to the hoist, the weight gain is normal.

B. The fuel assembly is bowed and contacting reactor vessel internals.

C. A grid-to-grid hang-up with an adjacent fuel assembly is occurring.

D. The camera has contacted the core support barrel.

Answer: A

Answer Explanation:

A. **Correct** - OI-25C references picking up the weight of the hoist box at about 126.5"

B. **Incorrect** - at this height, the fuel is inside the hoist box and above other fuel assemblies.

C. **Incorrect** - at this height, the fuel is inside the hoist box and above other fuel assemblies.

D. **Incorrect** - RFHM interlocks prevent this condition from occurring and the weight gain couldn't be attributed to camera interference.

Students are not expected to memorize specific heights associated with Refueling Machine operation but there is enough information in this question for the students to analyze each of the distractors and arrive at the correct answer, therefore higher cognitive

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Question 67 (Q14509)			
Topic:	RFHM operating characteristics		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.1 - Conduct of Operations • 2.1.41 - Knowledge of the refueling process		
RO Importance:	2.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-113-6-4-17		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	OI-25C, Refueling Machine		
Comments:	None		

68. 2.2 - Equipment Control

Which **ONE** of the following conditions would be considered preconditioning?

A. Performing a monthly STP on weekly basis for closer monitoring of valve performance.

B. Draining the Containment Sump prior to receipt of the alarm, 3 hrs before performing a timed stroke STP.

C. During performance of time stroke STP an error occurred using the stopwatch and performed the test again immediately.

D. Opening 8" CAC emergency outlet valve to lower a high Containment temperature condition, 3 hrs before the same emergency outlet valve time stroke STP

Answer: B

Answer Explanation:

A. **Incorrect** - EN-4-108 (3.13) specifically allows increased surveillance frequency

B. **Correct** - EN-4-108 (3.13) only allows operation if reasonably needed for plant evolution. Draining the Containment Sump prior to receipt of the alarm is not normal or reasonable

C. **Incorrect** - EN-4-108 (3.19) specifically allows retest for test equipment malfunction

D. **Incorrect** - EN-4-108 (3.13) specifically allows operation if reasonably needed for plant evolution

Each of these distractors would technically be preconditioning but the procedure makes allowances for only one. Students are not expected to memorize all potential preconditioning conditions but this question gives enough information that the students can analyze each situation and determine which one would be acceptable, therefore higher cognitive.

Question 68 (Q97178)			
Topic:	Pre-conditioning of SR components		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.2 - Equipment Control • 2.2.12 - Knowledge of the surveillance procedures		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EN-4-108, ASME In-service Testing Of Power-Operated Valves & Manual Valves		
Comments:	None		

69. 2.2 - Equipment Control

When preparing a tagout, which **ONE** of the following sets of values require the use of **two** isolation points, (1) Fluid Temperature (2) System Pressure

A. (1) > 200˚F, (2) > 500 psig
B. (1) > 200˚F, (2) > 250 psig
C. (1) > 120˚F, (2) > 250 psig
D. (1) > 120˚F, (2) > 500 psig

Answer: A

Answer Explanation:

A. **Correct** - - Per CNG-OP-1.01-1007 the following are considered for the use of double isolation:; Greater than to 200°F temperature; Greater than 500 psig;

B. **Incorrect** - - Per CNG-OP-1.01-1007 the following are considered for the use of double isolation: Greater than 200°F temperature; **Greater than 500 psig**

C. **Incorrect** - Per CNG-OP-1.01-1007 the following are considered for the use of double isolation: **Greater than 200°F temperature**; **Greater than 500 psig**

D. **Incorrect** - - Per CNG-OP-1.01-1007 the following are considered for the use of double isolation: **Greater than 200°F temperature**; Greater than 500 psig

;

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Question 69 (Q97179)			
Topic:	Knowledge of Safety Tagging procedures		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.2 - Equipment Control • 2.2.13 - Knowledge of safety and tagging clearance procedures		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	NO-1-112 Safety Tagging Guidance CNG-OP-1.01-1007, Clearance And Safety Tagging		
Comments:	None		

70. 2.2 - Equipment Control

Given the following:

- Unit 2 is in Mode 6, refueling has been completed
- SDC remains in service
- The reactor head stud bolts are being tensioned
- A penetration has a Containment closure deviation on file. It can be restored in 10 minutes
- The time to boil is 45 minutes.
- T.S 3.9.4/3.9.5 limitations are met.

What is the maximum amount of time the penetration may be left open?

A. 45 minutes

B. 7 days

C. Indefinitely (as long as mode 5 is not entered)

D. Indefinitely (as long as mode 4 is not entered)

Answer: D

Answer Explanation:

A. **Incorrect** - 45 minutes is the time to boil, as long as the penetration is able to be closed within TTB then not required to be shut

B. **Incorrect** - 7 days is requirement for STP O-55A periodicity while shutdown, , as long as the penetration is able to be closed within TTB then not required to be shut

C. **Incorrect** - Indefinitely is correct but mode change we are worried about is Mode 4 because the containment is required to have integrity vice closure in Mode 1-4

D. **Correct** - per NO-1-114 section 5.1.B "If a penetration can be closed in less than the TTB and T.S. 3.9.4/3.9.5 requirements, then there are no limits on the duration that the penetration is allowed to be open. As long as unit does not enter mode of applicability for containment integrity (Modes 1-4)

Question 70 (Q44860)			
Topic:	Limitations on closure openings		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.2 - Equipment Control • 2.2.14 - Knowledge of the process for controlling equipment configuration or status.		
RO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:			
Comments:	None		

71. 2.3 - Radiation Control

The following emergency situations may warrant an individual dose in excess of the established regulatory limit of 5 REM per year:

(1) Life Saving (voluntary)
(2) Facility Protection

Which **ONE** of the following represents the guidance for dose accumulated by an individual during these emergency situations?

A. (1) May exceed 25 REM;
(2) Limited to 10 REM

B. (1) May exceed 25 REM
(2) Limited to 25 REM

C. (1) Limited to 25 REM;
(2) Limited to 25 REM

D. (1) Limited to 10 REM,
(2) Limited to 10 REM

Answer: A

Answer Explanation:

A. **Correct** - Dose limits specified in ERPIP 831are: May exceed 25 REM for Lifesaving (voluntary) and limited to 10 REM for Facility Protection.

B. **Incorrect** - Dose limits specified in ERPIP 831are: May exceed **25 REM** for Lifesaving (voluntary) and limited to **10 REM** for Facility Protection.

C. **Incorrect** - Dose limits specified in ERPIP 831are: **May exceed 25 REM** for Lifesaving (voluntary) and limited to **10 REM** for Facility Protection.

D. **Incorrect** - Dose limits specified in ERPIP 831are: **May exceed 25 REM** for Lifesaving (voluntary) and limited to 10 REM for Facility Protection.

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Question 71 (Q92910)			
Topic:	Emergency dose limits		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.3 - Radiation Control • 2.3.4 - Knowledge of radiation exposure limits under normal or emergency conditions.		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(12)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	Used in the 8/2010 NRC exam		
Technical references:	ERPIP 831, Emergency Radiation Exposure Guidance.		
Comments:	None		

72. 2.3 - Radiation Control

You are signed onto the Ops Low Risk RWP #2 Task #2 for an entry into the Aux Bldg, with limits of 10 mrem dose and 80 mrem/hr does rate.

Which set of alarms setpoints would you expect to see when checking your EPD prior to entry?

A. 8 mrem 64 mrem/hr

B. 8 mrem 80 mrem/hr

C. 10 mrem 64 mrem/hr

D. 10 mrem 80 mrem/hr

Answer: B

Answer Explanation:

The alarms are set on the EPD to 80% of dose limit and at the dose rate limit. Therefore correct answer is 8 mrem & 80 mrem/hr

A. **Incorrect** – See explanation above

B. **Correct** - See explanation above

C. **Incorrect** – See explanation above

D. **Incorrect** – See explanation above

Question 72 (Q74533)			
Topic:	RWP use in emergency situations		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.3 - Radiation Control • 2.3.7 - Ability to comply with radiation work permit requirements during normal or abnormal conditions.		
RO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(12)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	RWP-2-2, Operations Activities		
Comments:	None		

73. 2.3 - Radiation Control

U-1 is operating normally at 100% when a S/G tube leak of approximately 50 GPM develops.

Which **ONE** of the following steps, directed by AOP-2A, Excessive Reactor Coolant Leakage, is performed to prevent the release of contamination?

A. Reduce turbine load to maintain T_{COLD} on program

B. Ensure no leakage into Component Cooling subsystem

C. Direct Rad Safety to survey Water Treatment Area

D. Borate to reduce $T_{AVE} < 537°F$

Answer: D

Answer Explanation:

A. **Incorrect** - Per AOP-2A: "This allows the operator to control the plant using an observable parameter while dropping load as temperature is being reduced to < 537°F. In the past the operator was trying to keep temperature near program which was constantly changing and had to be looked up. Controlling SG pressure in this band keeps the plant from swinging as power is fairly rapidly reduced."

B. **Incorrect** - step only performed if location of leak not identified. Information given is that a tube leak in excess of 1 charging pump therefore unit is tripped in Step VI.D and this step (VI.E.10) not performed

C. **Incorrect** - This step is in section V, RCS leakage within capacity of 1 charging pump. Information given states 15 gpm which exceeds capacity of charging pump with L/D in service. Therefore this step is not performed.

D. **Correct** - AOP-2A basis states: **"**the intent of this step is to reduce T_{AVE} to less than 537°F if possible before tripping the Reactor to minimize the possibility of lifting a Steam Generator Safety Valve. The setpoint of the lowest lifting main steam safety valve corresponds to a saturation temperature of approximately 537°F.

Question 73 (Q97180)			
Topic:	Control of contamination during a SGTR Event		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.3 - Radiation Control • 2.3.14 - Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(12)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EOP-6, Steam Generator Tube Rupture EOP-6, Steam Generator Tube Rupture Technical Basis Document		
Comments:	None		

74. 2.4 - Emergency Procedures / Plan

Unit 1 is operating at 60% power when a loss of 4KV Bus 13 occurs.

Which **ONE** of the following actions should be taken?

A. Bypass Condensate Precoat Filters and Demineralizers and verify or start 12 or 13 Condensate Pump

B. With concurrence of the SM/CRS trip the Reactor and Turbine then implement EOP-0.

C. Perform a power reduction to lower Condensate Header flow to below 8000 GPM

D. Isolate S/G Blowdown and verify 12 Heater Drain Pump is running.

Answer: C

Answer Explanation:

A. **Incorrect** - 12 and 13 Condensate Pumps are powered by 4KV Bus 13.

B. **Incorrect** - Trip criteria is provided in AOP-7I-1 and not met initially (<70%), following the guidance to reduce power as necessary to maintain Condensate header flow less than 8000 GPM. Tripping the Reactor and Turbine would be based on exceeding the trip criteria during the course of the power reduction.

C. **Correct** - AOP-7I-1, for the loss of 13 4KV Bus, states: **IF** Reactor power is less than 70% Reduce power as necessary to maintain Condensate header flow less than 8000 GPM **PER** OP-3, Normal Power Operation.

D. **Incorrect** - 12 Heater Drain Pump is powered by 4KV Bus 13.

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Question 74 (Q28551)			
Topic:	Response to loss of 13 4KV Bus		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.4 - Emergency Procedures / Plan • 2.4.11 - Knowledge of abnormal condition procedures.		
RO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power		
Comments:	None		

75. 2.4 - Emergency Procedures / Plan

(1) Where does the Operations Technical Advisor report to on declaration of a Fire and (2) what is the Operations Technical Advisor's function?

A. (1) Reports to the Control Room;
(2) Provides technical assistance regarding firefighting strategies

B. (1) Reports to the Control Room;
(2) Provides technical assistance regarding safe shutdown equipment

C. (1) Reports to the scene of the Fire;
(2) Provides technical assistance regarding safe shutdown equipment

D. (1) Reports to the scene of the Fire;
(2) Provides technical assistance regarding firefighting strategies

Answer: C

Answer Explanation:

A. **Incorrect** - Per SA-1-101 section 5.1.D: The Operations Technical Advisor shall be a licensed operator who **responds with the Fire Brigade** to provide technical assistance to the Fire Brigade Leader regarding **safe shutdown equipment functions**.

B. **Incorrect** - Per SA-1-101 section 5.1.D: The Operations Technical Advisor shall be a licensed operator who **responds with the Fire Brigade** to provide technical assistance to the Fire Brigade Leader regarding safe shutdown equipment functions.

C. **Correct** - Per SA-1-101, Section 4.6.A. "The Operations Technical Advisor shall be responsible to, in response to a fire, provide information on safe shutdown equipment to the Fire Brigade Leader and report the status of conditions in the area to the Control Room."
Per SA-1-101 section 5.1.D: The Operations Technical Advisor shall be a licensed operator who responds with the Fire Brigade to provide technical assistance to the Fire Brigade Leader regarding safe shutdown equipment functions.

D. **Incorrect** - Per SA-1-101 section 5.1.D: The Operations Technical Advisor shall be a licensed operator who responds with the Fire Brigade to provide technical assistance to the Fire Brigade Leader regarding **safe shutdown equipment functions**.

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Question 75 (Q50730)			
Topic:	OTA responsibilities		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.4 - Emergency Procedures / Plan • 2.4.25 - Knowledge of fire protection procedures.		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	SA-1-101, Fire Fighting		
Comments:	None		

CALVERT CLIFFS NUCLEAR POWER PLANT

2014 NRC

INITIAL LICENSED

OPERATOR

SRO WRITTEN EXAM

KEY

Revision 1 (Incorporates Ops Reviewer Comments)

- #77 – Minor editorial correction to the stem of the question
- #79 – Minor editorial correction to the stem of the question
- #82 – Minor Changes to answers A & B (unaffected CEA's)
- #83 – Minor editorial correction to the stem of the question
- #85 – Changes to stem and distractors to eliminate ambiguity
- #86 – Changes to stem and distractors to eliminate ambiguity
- #87 – Minor editorial correction to the stem of the question
- #90 – Minor editorial correction to the stem of the question
- #91 – Changed the fault to LRNI (Q19 had WRNI Power Supply)
- #98 – Minor editorial correction to the stem of the question
- #99 – Minor editorial correction to the stem of the question
- #100 – Changed ERPIP form titles to new standard
- Numerous changes to distractor order to equalize correct answers: 6-A, 7-B, 6-C, 6-D

Revision 2 (Incorporates Collegial Review Comments)

- #79 – Minor change to stem
- #83 – Added "Using Provided Reference"
- #87 – Reworded stem to clarify
- #90 – Minor changes to wording of stem and swap (1) & (2) to make clearer
- #91 – Minor change in stem to make clearer

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Revision 3 (Incorporates Exam Validator Comments)

- #82 – Replaced question with another that matched the K/A
- #86 – Minor change to distractors A & C to make clearer

Revision 4 (Incorporates 2nd Round Exam Validator Comments)

- #79 – Minor style change to clarify 2 parts of question/answer
- #87 – Changed N-16 to MSLRM in stem for better operational validity
- #93 – Minor editorial change to stem
- #100 – Added using provided reference (EAL Hot Chart)

Revision 5 (Incorporates Fleet Reviewer and NRC pre-review Comments)

- #76 – Changed stem to cause more diagnosis based on NRC comment about LOD
- #78 – Minor editorial changes to distractors A, B, & C
- #79 – Added explanation as to why this is as SRO question
- #81 – Minor editorial change to stem, added "implemented on U-1"
- #82 – Rewrote question based on NRC feedback about SRO question
- #87 – Minor editorial changes to stem based on NRC feedback
- #90 – Editorial change to stem (bold "**all**") and added "potential"
- #93 – Minor editorial change to stem to clarify K/A match

76. 022 - Loss of Rx Coolant Makeup

Using provided references:

U-1 is in Mode 3 and cooling down to Mode 5 for a refueling outage. The following conditions exist:

- The RCS has been borated to refueling boron concentration.
- CVCS is aligned for auto makeup to the VCT.

Assuming the demineralized water flow transmitter feeding the MAKEUP WTR FLOW CONTR (FIC-210X) was inadvertently isolated sometime in the past 2 hours. Which **ONE** of the following describes: (1) The effect this failure has on the RCS and (2) What direction would you give to RO?

A. (1) RCS boron concentration would be lower than the last sample
(2) Stop cooldown, stop auto makeup to VCT, direct Chemistry to sample the RCS

B. (1) RCS boron concentration would be lower than the last sample
(2) Continue cooldown, immediately commence fast boration to restore SDM

C. (1) RCS boron concentration would be higher than the last sample
(2) Stop cooldown, stop auto makeup to VCT, direct Chemistry to sample the RCS

D. (1) RCS boron concentration would be higher than the last sample
(2) Continue cooldown, dilute the RCS to prevent exceeding max boron concentration.

Answer: A

Answer Explanation:

A. **Correct** – Isolating the DI flow transmitter would cause indication of 0 gpm which would cause controller to fully open causing excessive DI flow which would dilute the RCS but 1 hour of excessive DI flow would not be expected to cause a violation of SDM since stem states that all BA has been added up to refueling concentration therefore stopping auto M/U and sample RCS are appropriate. With no makeup to VCT the cooldown should also be stopped.

B. **Incorrect** - SDM not violated. Refueling boron concentration is 2505 ppm and most restrictive SDM in mode 5 is 1517 ppm. DI M/U controller is set for 100 gpm and failing high would cause it to put out max of DI transfer pump. To dilute from 2500 ppm to 1500 ppm is 37000 gallons of DI which would require >300gpm of DI flow (max is <200gpm) for entire 120 mins.

C. **Incorrect** - RCS concentration would be lower not higher

D. **Incorrect** - RCS concentration would be lower not higher

Question 76 (Q97190)			
Topic:	SRO Only RCMU controller failure		
Tier/Group:	1/1		
K/A Info:	022 - Loss of Rx Coolant Makeup • AA2 - Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: • AA2.03 - Failures of flow control valve or controller		
SRO Importance:	3.6		
Proposed references to be provided to applicant:	NEOP-13 P.22 - 29 and OI-2B Figures 1-4		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	AOP-1A, Inadvertent Boron Dilution		
Comments:	None		

77. 040 - Steam Line Rupture - Excessive Heat Transfer

Given U-2 with the following conditions:

- RCS pressure is 1545 PSIA slowly lowering
- Containment pressure is 3.1 PSIG and slowly rising
- 21 & 22 S/G pressures are approximately 720 PSIA and slowly lowering
- CRO was delayed in shutting MSIV's, CRS has directed them to be shut

Which **ONE** of the following describes (1) The correct Optimal Recovery Procedure to implement and (2) The first Block Step to be assigned to RO/CRO from the selected procedure?

A. (1) Implement EOP-5, Loss of Coolant Accident;
(2) Monitor RCS Depressurization

B. (1) Implement EOP-5, Loss of Coolant Accident;
(2) Depressurize the RCS to Reduce Subcooling and Maintain Pressurizer Level

C. (1) Implement EOP-4, Excess Steam Demand Event;
(2) Monitor ESFAS Actuation

D. (1) Implement EOP-4, Excess Steam Demand Event;
(2) Maintain RCS Subcooling and Pressurizer Level

Answer: C

Answer Explanation:

A. **Incorrect** - The parameters provided support diagnosis of a steam line break in the Containment, rather than a LOCA, as evidenced by a subcooled margin of 94°F. Monitoring RCS depressurization would be the correct block step selection if the event were a LOCA.

B. **Incorrect** - The parameters provided support diagnosis of a steam line break in the Containment, rather than a LOCA, as evidenced by a subcooled margin of 94°F. Depressurize the RCS to Reduce Subcooling and Maintain Pressurizer Level would be the correct block step selection (but not the first one) if the event were a LOCA.

C. **Correct** - The parameters provided support diagnosis of a steam line break in the Containment warranting implementation of EOP-4, Excess Steam Demand Event. Monitor ESFAS Actuation is the first step assigned to one of the board operators.

D. **Incorrect** - The parameters provided support diagnosis of a steam line break in the Containment warranting implementation of EOP-4, Excess Steam Demand Event. This block step provides guidance for maintaining RCS subcooling and pressurizer level and is situated after affected S/G isolation in the procedure.

Question 77 (Q97155)			
Topic:	Diagnose EOP-4 entry and select 1st block step		
Tier/Group:	1/1		
K/A Info:	040 - Steam Line Rupture - Excessive Heat Transfer • 2.4 - Emergency Procedures / Plan: • 2.4.1 - Knowledge of EOP entry conditions and immediate action steps.		
SRO Importance:	4.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	EOP-0, POST-Trip Immediate Actions EOP-4, Excess Steam Demand Event		
Comments:	None		

78. 055 - Station Blackout

A station blackout has occurred, EOP-0, Post-Trip Immediate Actions has been completed and the appropriate Optimal Recovery procedure implemented.

Which **ONE** of the following statements meet the procedure requirement(s) to exit this procedure?

A. The Safety Function Status Check Final Acceptance Criteria are being met permitting transition to OP-4 "Plant Shutdown from Power Operation to Hot Standby"

B. Both 4KV Vital buses and associated load centers have been re-energized permitting transition to EOP-2 "Loss of Offsite Power / Loss of Forced Circulation"

C. One 4KV Vital bus and associated load centers has been re-energized permitting transition to OP-4 "Plant Shutdown from Power Operation to Hot Standby"

D. The Tech Support Center determines Final Safety Function Acceptance Criteria is too conservative and recommends transition to OP-4 "Plant Shutdown from Power Operation to Hot Standby" and AOP-3F "Loss Of Offsite Power While In Modes 3, 4, 5 Or 6" in parallel

Answer: A

Answer Explanation:

A. **Correct** - All of the Optimal Recovery Procedure require the Safety Function Status Check Final Acceptance Criteria are met. Additionally, EOP-2 may be an acceptable Optimal Recovery Procedure to transition to.

B. **Incorrect** - All of the Optimal Recovery Procedure require the Safety Function Status Check Final Acceptance Criteria are met. Information is not provided on the status of meeting the Safety Function Status Check Final Acceptance Criteria.

C. **Incorrect** - All of the Optimal Recovery Procedure require the Safety Function Status Check Final Acceptance Criteria are met. Information is not provided on the status of meeting the Safety Function Status Check Final Acceptance Criteria.

D. **Incorrect** - All of the Optimal Recovery Procedure require the Safety Function Status Check Final Acceptance Criteria are met. Implementing the Tech Support Center's recommendations is not in accordance with the exit strategies outlined in EOP-7 and would require a procedure deviation be authorized by the Shift Manager

Question 78 (Q28785)			
Topic:	Exit EOP-7		
Tier/Group:	1/1		
K/A Info:	055 - Station Blackout • 2.1 - Conduct of Operations: • 2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.		
SRO Importance:	4.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	EOP-7, Station Blackout		
Comments:	None		

79. 062 - Loss of Nuclear Service Water

Unit-2 is at 100% power with the following indications:

- Service Water (SRW) Head Tank low level alarms
- Turbine Bldg sump alarm on 2T22
- Both SRW Head Tanks levels are ≈40 inches and lowering at ≈8 inches/min

After the correct AOP is entered which **ONE** of the following (1) action is required and (2) why is this action performed?

A. (1) Trip U-2 Reactor and turbine
(2) Reduce heat load to keep CAC's and D/G's available

B. (1) Shut Turbine Bldg. SRW Isolation Valves
(2) Isolate leak to keep CAC's and D/G's available

C. (1) Open SRW Head tank level control bypass valves
(2) To prevent SRW pumps from cavitating

D. (1) Commence power reduction per OP-3 "Normal Power Operation"
(2) Minimize plant transient if trip is required

Answer: B

Answer Explanation:

A. **Incorrect** - Tripping the reactor is a required step but reason is to prevent damage to components cooled by SRW, not to keep CAC's and D/G's available since TB should be isolated from AB before tripping reactor.

B. **Correct** - Indications given are rapidly lowering with indication of system rupture. Shutting TB isolation is step V.D.2.a and basis is to separate non safety related TB from safety related AB where CAC's and D/G's are supplied from.

C. **Incorrect** - With rapidly lowering SRW Head Tank opening level control bypass is unnecessary because level will be empty by the time this action can be completed and excessive leakage flow is continuing

D. **Incorrect** - Power reduction may be performed but intent is to reduce heat load on SRW system.

SRO question because the student must determine correct AOP and the proper section of that AOP to determine correct actions. Also the student must determine the reason for performing this step.

Question 79 (Q97194)			
Topic:	Actions necessary for a leak on 21 SRW header		
Tier/Group:	1/1		
K/A Info:	062 - Loss of Nuclear Service Water • AA2 - Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: • AA2.03 - The valve lineups necessary to restart the SWS while bypassing the portion of the system causing the abnormal condition		
SRO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A - new question		
Technical references:	AOP-7B, Loss of Service Water AOP-7B, Loss of Service Water Technical Basis		
Comments:	None		

80. CE/E02 - Reactor Trip - Stabilization - Recovery

A low flow Reactor trip has just occurred due to 11A RCP experiencing a locked rotor. The following conditions also exist during EOP-0:

- TBVs modulating RCS temperature in normal range
- 11 SG level is -80 inches and rising
- 12 SG level is -120 inches and slowly rising
- 11B RCP Bentley Nevada indicates 0 mils, the OK light is out and the Danger light is lit
- RCS pressure is 2000 PSIA and rising
- PZR level is 107 inches and slowly rising

Which **ONE** of the following EOPs should be implemented upon exiting EOP-0?

A. EOP-1, Reactor Trip

B. EOP-2, Loss of Offsite Power/Loss of Forced Circulation

C. EOP-6, Steam Generator Tube Rupture

D. EOP-8, Functional Recovery Procedure

Answer: A

Answer Explanation:

The pump configuration of one RCP in 11 loop and two RCPs in 12 loop have created a S/G level mismatch.

The vibration information given for the 11B RCP is trip criteria for the pump in OI-1A. The operator would be expected to trip the RCP during EOP-0, yielding a final pump configuration of no RCPs in 11 loop and two RCPs in 12 loop.

A. **Correct** - The EOP that would be implemented next would be EOP-1 since the only complications were the RCPs before and after the trip and all safety functions are met.

B. **Incorrect** - EOP-2 is implemented during a loss of **all** forced circulation. Since both 12 loop RCPs are still operating, natural circulation does not exist and EOP-2 is not desired.

C. **Incorrect** - SG level mismatch is due to RCP configuration and not a tube leak. 'C' could be picked if the diagnostic flowchart was improperly evaluated as HR not met with feed flow and RCPs tripped.

D. **Incorrect** - The conditions stated do not lead to meeting any of the entry criteria for implementation of the Functional Recovery Procedure.

Question 80 (Q95235)			
Topic:	EOP Transition with both 11A/11B RCP secured		
Tier/Group:	1/1		
K/A Info:	CE/E02 - Reactor Trip - Stabilization - Recovery • 2.4 - Emergency Procedures / Plan: • 2.4.31 - Knowledge of annunciator alarms, indications, or response procedures		
SRO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Demonstrate an understanding of the Strategy, Basis, and Operator actions of EOP-0.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on a previous NRC exam		
Technical references:	OI-1A Reactor Coolant System and Pump Operations; 1C06-ALM RCS Control Alarm Manual; EOP-0 Post-Trip Immediate Actions		
Comments:	None		

81. CE/E06 - Loss of Feedwater

U-1 has experienced a reactor trip. Given the following conditions at the completion of EOP-0:

- Loss of Offsite Power occurred
- 11 & 24 4KV buses faulted
- 12 ADV cannot be opened
- 11 AFW Pump OOS for overhaul
- 12 AFW Pump tripped and cannot be reset

Which **ONE** of the following procedures should be implemented on U-1?

A. EOP-2, Loss of Offsite Power/Loss of Forced Circulation

B. EOP-3, Loss of All Feedwater

C. EOP-4, Excess Steam Demand Event

D. EOP-8, Functional Recovery Procedure

Answer: B

Answer Explanation:

A. **Incorrect** - May be chosen if it is not recognized 13 & 23 AFW power supplies are deenergized due to loss of the specified buses

B. **Correct** - Conditions presented result in a Loss of All Feedwater condition. EOP-3, Loss of All Feedwater is written to address a Loss of Offsite Power condition coincident with the Loss of All Feedwater.

C. **Incorrect** - Plausible since 12 ADV not opening may cause safety valves to open which could cause an ESDE given an equipment malfunction.

D. **Incorrect** - Plausible if indications provided are misinterpreted as indicating multiple events (ESDE and LOAF) are occurring. IF this were true, EOP-8, Functional Recovery Procedure, would be appropriate.

Question 81 (Q97156)			
Topic:	Select the appropriate EOP for stated conditions		
Tier/Group:	1/1		
K/A Info:	CE/E06 - Loss of Feedwater • EA2 - Ability to determine and interpret the following as they apply to the (Loss of Feedwater): • EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations		
SRO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	EOP-0, Post-Trip Immediate Actions and associated Technical Basis document.		
Comments:	None		

82. 003 - Dropped Control Rod

Given U-1 operating at 50% Beginning of Cycle, a malfunction occurs with the following indications:

- Tcold 537°F
- Gross electrical output dropped by 40 MWe
- Linear Range "A" NI Power 47%
- Linear Range "C" NI Power 43%

Which **ONE** of the following is (1) correct procedure to implement and (2) action required by this procedure?

A. (1) AOP-1B "CEA MALFUNCTION"
(2) Lower turbine load to restore Tcold to program

B. (1) AOP-7K "OVERCOOLING EVENT IN MODE 1 OR 2"
(2) Lower turbine load to restore Tcold to program

C. (1) AOP-1B "CEA MALFUNCTION"
(2) Trip U-1 Reactor due to exceeding trip criteria

D. (1) AOP-7K "OVERCOOLING EVENT IN MODE 1 OR 2"
(2) Shut affected Turbine Bypass Valve upstream isolation valve

Answer: A

Answer Explanation:

A. **Correct** – Indications given in stem (Tcold 3°F low, 4% differential in power indications) indicate a dropped CEA so AOP-1B is correct procedure to implement. Lowering turbine load is the correct action to take (no trip criteria met yet)

B. **Incorrect** – All of the indications except the difference in NI channel power indicate an overcooling event but all indication are consistent with a dropped CEA.

C. **Incorrect** – Indications given in stem (Tcold 3°F low, 4% differential in power indications) indicate a dropped CEA so AOP-1B is correct procedure to implement. Lowering turbine load is the correct action to take (no trip criteria met yet)

D. **Incorrect** – All of the indications except the difference in NI channel power indicate an overcooling event but all indication are consistent with a dropped CEA. If student decides an overcooling event has occurred isolating a leaking TBV would be correct.

Question 82 (Q40688)			
Topic:	Response to a dropped CEA from 50% power		
Tier/Group:	1/2		
K/A Info:	003 - Dropped Control Rod • AA2 - Ability to determine and interpret the following as they apply to the Dropped Control Rod: • AA2.03 - Dropped rod, using in-core/ex-core instrumentation, in-core or loop temperature measurements.		
SRO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	AOP-1B, CEA Malfunction AOP-1B Basis		
Comments:	None		

83. 024 - Emergency Boration

Using provided reference:

1-CVC-MOV-504 "RWT CHG PP SUCT" is tagged shut for maintenance when a fault initiates on 14 4KV bus. U-1 is manually tripped. You have been designated to review the TS and TRM implications from the events in the trip.

Which **ONE** of the following describes the actions required, if any, to satisfy the TRM for Boration Flow Paths?

A. No actions are required.

B. Log entry into and request a Functional Assessment for 15.0.3.

C. Log that required boron injection flow path needs to be restored within 7 days.

D. Log that required boron injection flow path needs to be restored within 72 hours.

Answer: D

Answer Explanation:

A. **Incorrect** - TRM 15.1.2.A, requiring a flow path to be restored within 72 hours, is applicable.

B. **Incorrect** - With MOV-504 tagged shut, the RWT is not available as a boration flowpath. When 14 4KV bus was lost, 12 Charging Pump, 12 BA Pump, and CVC-514 were lost due to no power. Two boration sources were available (11 BAST through MOV-509 and 12 BAST through MOV-508), but 11 and 13 Charging Pump were not powered from separate busses. Therefore TRM 15.1.2.A applied, requiring a flow path to be restored within 72 hours.

C. **Incorrect** - With MOV-504 tagged shut, the RWT is not available as a boration flowpath. When 14 4KV bus was lost, 12 Charging Pump, 12 BA Pump, and CVC-514 were lost due to no power. Two boration sources were available (11 BAST through MOV-509 and 12 BAST through MOV-508), but 11 and 13 Charging Pump were not powered from separate busses. Therefore TRM 15.1.2.A applied, requiring a flow path to be restored within 72 hours.

D. **Correct** - With MOV-504 tagged shut, the RWT is not available as a boration flowpath. When 14 4KV bus was lost, 12 Charging Pump, 12 BA Pump, and CVC-514 were lost due to no power. Two boration sources were available (11 BAST through MOV-509 and 12 BAST through MOV-508), but 11 and 13 Charging Pump were not powered from separate busses. Therefore TRM 15.1.2.A applied, requiring a flow path to be restored within 72 hours.

Question 83 (Q92671)			
Topic:	TRM implications of loss of power		
Tier/Group:	1/2		
K/A Info:	024 - Emergency Boration • 2.2 - Equipment Control: • 2.2.40 - apply Technical Specifications for a system.		
SRO Importance:	4.7		
Proposed references to be provided to applicant:	NO-TRM, Technical Requirements Manual pages1,12-17		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	NO-TRM, Technical Requirements Manual		
Comments:	None		

84. 037 - Steam Generator Tube Leak

Given the following:

- A Steam Generator Tube Rupture has occurred
- SIAS has actuated
- RCS pressure has stabilized at 1200 PSIA
- The affected S/G has been isolated
- You direct the RO to perform the next step of the Optimal Recovery procedure "DEPRESSURIZE THE RCS TO REDUCE SUBCOOLING AND MAINTAIN PRESSURIZER LEVEL"

Which **ONE** of the following describes the optimal RCS Pressure/subcooling band you would direct the RO to maintain per the Optimal Recovery Procedure Basis and why?

A. RCS pressure above RCP curve but less than 60°F subcooling;
Minimize leakage from RCS to S/G and ensure forced circulation maintained

B. RCS pressure above RCP curve but less than 140°F subcooling;
Ensure forced circulation maintained and subcooled fluid surrounding the core

C. 25 to 140°F subcooling margin;
Ensures RCS remains subcooled and prevents (PTS) pressurized thermal shock

D. 25-50°F subcooling margin;
Minimize leakage from RCS to S/G

Answer: A

Answer Explanation:

A. **Correct** - Keeping Subcooling in this band maintains RCP's running and minimizes RCS leak rate

B. **Incorrect** - Because it allows high subcooling which allows higher RCS leak rates

C. **Incorrect** - Because with given conditions one RCP in each loop is running and 25°F subcooling is below RCP pump curve

D. **Incorrect** - 25-140°F of subcooling represents the limits specified in EOP-6, however, the basis emphasizes maintaining subcooling low in band to minimize RCS leak rate

SRO only justified per 10CFR55.43(b)(5) per NRC guidance document for SRO questions. **Does the question involve one or more of the following? Recalling what strategy or action is written into a plant procedure, including when the strategy or action is required.**

Question 84 (Q97186)			
Topic:	Maintaining SCM with a SGTR		
Tier/Group:	1/2		
K/A Info:	037 - Steam Generator Tube Leak • AA2 - Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: • AA2.16 - Pressure at which to maintain RCS during S/G cooldown.		
SRO Importance:	4.3		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	EOP-6, Steam Generator Tube Rupture; EOP-6, Steam Generator Tube Rupture Technical Basis		
Comments:	None		

85. CE/E09 - Functional Recovery

A reactor trip and SIAS has occurred on U-1. While performing EOP-0, the RO and CRO provide the following information to the CRS:

- 2 CEAs failed to insert and boration has commenced
- Pressurizer level is 70 inches and continuing to lower with all Charging pumps running and Letdown isolated
- RCS pressure is 1500 PSIA and steady
- T_{COLD} is 532°F
- Both S/G levels are -20 inches and rising towards 0 inches with AFW flow
- Both S/G pressures are 900 PSIA and steady
- Containment pressure is 0.8 PSIG and steady; alternate actions performed
- Containment temperature is 119°F and steady
- S/G Blowdown and Main Vent RMS are in alarm

From the information provided, which safety function report is correct, and which procedure would you implement?

A. Reactivity Control, Core/RCS Heat Removal, and Containment Environment are Complete, implement EOP-6

B. Reactivity Control, Core/RCS Heat Removal, and Containment Environment are Complete, implementEOP-8

C. Rad Levels External to Containment and RCS Pressure/Inventory Control Cannot Be Met, implement EOP-6

D. Rad Levels External to Containment and RCS Pressure/Inventory Control Cannot Be Met, implement EOP-8

Answer: D

Answer Explanation:

A. **Incorrect** - RC is MET (boration in progress), HR is MET and CE is NOT MET (Containment Pressure). EOP-8 would be the appropriate procedure to enter because multiple events are in progress.

B. **Incorrect** - RC is MET (boration in progress), HR IS MET and CE is NOT MET (Containment Pressure)

C. **Incorrect** - EOP-8 would be the appropriate procedure to enter because multiple events are in progress.

D. **Correct** - Safety Function status reports are correct for stated conditions. EOP-8 would be the appropriate procedure to enter because multiple events are in progress.

Question 85 (Q97157)			
Topic:	EOP-8 Parameters and logic		
Tier/Group:	1/2		
K/A Info:	CE/E09 Functional Recovery • 2.4 - Emergency Procedures / Plan: • 2.4.21 - Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.		
SRO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	EOP-0, Post-Trip Immediate Actions; EOP-8, Functional Recovery Procedure		
Comments:	None		

86. 026 - Containment Spray System

U-2 has the following indications:

- A Design Basis Loss of Coolant Accident occurred 45 minutes ago
- 21 & 23 HPSI Pump amps are fluctuating
- HPSI flow has been throttled equally among the four headers to the minimum allowed

Which **ONE** of the following describes:
(1) What actions are to be taken IAW applicable procedure for conditions above?
(2) Impact of these actions?

A. (1) Stop both HPSI Pumps and implement EOP-8
(2) Potential to exceed maximum containment pressure limit

B. (1) Stop both CS Pumps and implement EOP-8
(2) Potential to exceed maximum containment pressure limit

C. (1) Stop both HPSI Pumps and implement EOP-8
(2) Potential to raise NPSH to CS pumps

D. (1) Stop both CS Pumps and implement EOP-8
(2) Potential to raise NPSH to HPSI pumps

Answer: D

Answer Explanation:

A. **Incorrect** - The four Containment Air Coolers supply a 100% redundancy to the Containment Spray System in supplying 100% of the capacity required in cooling the Containment. EOP-5, Loss of Coolant Accident, directs securing both CS Pumps.

B. **Incorrect** - The four Containment Air Coolers supply a 100% redundancy to the Containment Spray System in supplying 100% of the capacity required in cooling the Containment. Given that HPSI has been throttled to the minimum allowed flow, securing the CS Pumps is the next step in the procedure followed by checking the HPSIs for acceptable performance and then implementation of EOP-8.

C. **Incorrect** - EOP-5, Loss of Coolant Accident, **directs securing both CS pumps** then checking the performance of the HPSI Pumps. Securing both HPSI pumps is directed later in procedure if securing CS PP's did not cause SAT HPSI operation.

D. **Correct** - EOP-5, Loss of Coolant Accident, directs securing both CS pumps then checking the performance of the HPSI Pumps.

Question 86 (Q97191)			
Topic:	Diagnosing/responding		
Tier/Group:	2/1		
K/A Info:	026 - Containment Spray System (CSS) • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.07 - Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding), or sump level below cutoff (interlock) limit		
SRO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	EOP-5 EOP-5 Basis Document		
Comments:	None		

87. 039 - Main Steam and Reheat

U-2 was operating steady state at 100% when 21 Main Steam Line Radiation Monitor comes into alarm and is validated. AOP-2A "Excessive Reactor Coolant Leakage" has been implemented.

Which combination below, of (1) plant condition and (2) action from appropriate section of AOP-2A is correct?

A. (1) Pressurizer level rising slowly with L/D at minimum
(2) Borate to reduce T_{AVE} to < 537°F

B. (1) Pressurizer level lowering until 2nd Charging Pump starts
(2) Borate to reduce T_{AVE} to < 537°F

C. (1) Pressurizer level rising slowly with L/D at minimum;
(2) Commence an expeditious power reduction; maintain S/G pressure 800-825 PSIA

D. (1) Pressurizer level lowering until 2nd Charging Pump starts
(2) Commence an expeditious power reduction; maintain S/G pressure 800-825 PSIA

Answer: B

Answer Explanation:

A. **Incorrect** – PZR level rising with L/D at min indicates < capacity of 1 charging pump. Section V of AOP-2A does not direct borating to 537°F T_{AVE}.

B. **Correct** – PZR level lowering with L/D at minimum indicates > capacity of 1 charging pump. Sect VI of AOP-2A directs borating to 537°F T_{AVE}.

C. **Incorrect** - PZR level rising with L/D at min indicates < capacity of 1 charging pump. Section V of AOP-2A does not direct maintaining S/G press 800-825 PSIA.

D. **Incorrect** - PZR level lowering with L/D at minimum indicates > capacity of 1 charging pump. Sect VI of AOP-2A directs borating to 537°F T_{AVE} not an expeditious power reduction.

Question 87 (Q97182)			
Topic:	Steam Generator Tube Leak response		
Tier/Group:	2/1		
K/A Info:	039 - Main Steam and Reheat • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.03 - Indications and alarms for Main Steam and area radiation monitors (during SGTR)		
SRO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	AOP-2A, Excessive Reactor Coolant Leakage		
Comments:	None		

88. 061 - Auxiliary/Emergency Feedwater

Unit 1 is at 75% power when two of three (2 of 3) condensate pumps trip. The TBO reports the condensate header has ruptured, spraying water near the Condensate Booster Pumps.

Which **ONE** of the following strategies is employed to combat this condition?

A. Perform a Rapid Power Reduction to remove U-1 from service, secure Main Feed and Condensate systems and initiate AFW flow at <1% reactor power.

B. Secure remaining Condensate Pump, trip reactor prior to RPS Low S/G Level auto trip, implement EOP-0.

C. Trip the Reactor, verify Reactivity Control, secure the Main Feed and Condensate systems, start an AFW pump, and continue with EOP-0.

D. Commence a Rapid Power Reduction to lower Condensate Header flow to < 8000 GPM and direct TBO to isolate leak if possible.

Answer: C

Answer Explanation:

A. **Incorrect** - AOP-3G directs tripping reactor, performing reactivity control safety function, securing the Main feed and condensate systems, starting an AFW pump and continuing with EOP-0.

B. **Incorrect** - AOP-3G directs tripping reactor first and performing reactivity control safety function then securing main feed and condensate systems, starting an AFW pump and continuing with EOP-0.

C. **Correct** - IAW AOP-3G-1, Sect. VIII, these actions are all required in this order.

D. **Incorrect** - AOP-3G directs tripping reactor first and performing reactivity control safety function then securing main feed and condensate systems, starting an AFW pump and continuing with EOP-0.

Question 88 (Q28769)			
Topic:	AOP-3G Actions		
Tier/Group:	2/1		
K/A Info:	061 - Auxiliary/Emergency Feedwater • 2.4 - Emergency Procedures / Plan • 2.4.6 - Knowledge of EOP mitigation strategies		
SRO Importance:	4.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	EOP-3, Loss of All Feedwater		
Comments:	None		

89. 064 - Emergency Diesel Generator

Using provided reference(s):

Given the following conditions following a Loss of Offsite Power (LOOP):

- 1A, 1B, & 0C D/G's cannot be aligned due to 11 and 14 4KV buses faulted
- Repairs to 11 4KV Bus are projected to take 6 hours
- Repairs to 14 4KV Bus are projected to take 30 minutes
- Unit-1 RCS temperature is currently at 350°F
- A plant heatup to hot standby was in progress on Unit-1 prior to LOOP
- 4KV Bus 21 has been repowered from the 2A DG
- 4KV Bus 24 can be repowered from the 0C DG as soon as the disconnects are shifted

Which **ONE** of the following EAL classifications is appropriate for the stated conditions?

A. General Emergency

B. Site Area Emergency

C. Alert

D. Unusual Event

Answer: B

Answer Explanation:

A. **Incorrect** - With 14 4KV Bus expected to return in 0.5 hours conditions are met for a SAE not a General Emergency. A General Emergency declaration would be appropriate if Unit-1 were to be without power to a vital 4KV Bus for greater than 4 hours.

B. **Correct** - With 14 4KV Bus expected to return in 0.5 hours conditions are met for declaration of a Site Area Emergency for Unit-1.

C. **Incorrect** - Unit-2 is one power supply loss away from a Station Blackout which meets Alert declaration criteria. However, there is no power to either Unit-1 4KV bus therefore declaration of a Site Area Emergency is appropriate,

D. **Incorrect** - Unit-2 is meeting the conditions for Alert declaration. If the OC DG were pre-aligned to 24 4KV Bus, Unit-2 would be meeting the criteria for an Unusual Event declaration. However, Unit-1 conditions require declaring a SAE due to higher EAL category.

Question 89 (Q15827)			
Topic:	Evaluate 4KV Bus status for EAL Declaration purposes		
Tier/Group:	2/1		
K/A Info:	064 - Emergency Diesel Generator • 2.4 - Emergency Procedures / Plan • 2.4.41 – Knowledge of the EAL thresholds and classifications		
SRO Importance:	4.6		
Proposed references to be provided to applicant:	EAL Hot and Cold Charts		
Learning Objective:	Given a specific set of plant conditions, evaluate those conditions against the Emergency Action Levels and classify the emergency.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	ERPIP-3.0, Immediate Actions		
Comments:	None		

90. 078 - Instrument Air

U-1 is in Mode 5 with SDC in-service when a malfunction occurs on the in-service instrument air dryer.

Which **ONE** of the following describes (1) which procedure should be implemented to address **all** plant effects of this malfunction and (2) potential impacts to Shutdown Cooling from this malfunction?

A. AOP-3B, Loss of SDC;
SI-306-CV fails open, SI-657-CV fails closed

B. AOP-7D, Loss of Instrument Air;
SI-306-CV fails open, SI-657-CV fails closed

C. AOP-3B, Loss of SDC;
SI-306-CV fails closed, SI-657-CV fails open

D. AOP-7D, Loss of Instrument Air;
SI-306-CV fails closed, SI-657-CV fails open

Answer: B

Answer Explanation:

A. **Incorrect** - Loss of Instrument Air causes SI-306-CV to fail open and SI-657-CV fails shut. AOP-3B has steps for most malfunctions that could affect SDC (loss of power, leaks, etc.) but not a loss of instrument air. Therefore AOP-7D is required to correct an air dryer malfunction.

B. **Correct** - Loss of Instrument Air causes SI-306-CV to fail open and SI-657-CV fails shut. AOP-3B has steps for most malfunctions that could affect SDC (loss of power, leaks, etc.) but not a loss of instrument air. Therefore AOP-7D is required to correct an air dryer malfunction.

C. **Incorrect** - Loss of Instrument Air causes SI-306-CV to fail open and SI-657-CV fails shut. AOP-3B has steps for most malfunctions that could affect SDC (loss of power, leaks, etc.) but not a loss of instrument air. Therefore AOP-7D is required to correct an air dryer malfunction.

D. **Incorrect** - Loss of Instrument Air causes SI-306-CV to fail open and SI-657-CV fails shut. AOP-7D is required to correct an air dryer malfunction.

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Question 90 (Q97189)			
Topic:	Determine Air Dryer malfunction effects and correct AOP		
Tier/Group:	2/1		
K/A Info:	078 - Instrument Air • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.01 - Air dryer and filter malfunctions		
SRO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	AOP-3B, Abnormal Shutdown Cooling Conditions AOP-7D, Loss of Instrument Air		
Comments:	None		

91. 015 - Nuclear Instrumentation

U-1 is operating at 100% power when Channel B LRNI Power Summer failure occurs. Associated RPS Trip Units have been bypassed within required time.

Which **ONE** of the following describes (1) which RPS T/U's are affected and (2) how long until Channel B must be restored or additional actions must be taken.

A. (1) 1-High Power, 2-High SUR, 7-TM/LP, 8-APD, 10-Loss of Load
(2) 24 hours

B. (1) 1-High Power, 2-High SUR, 7-TM/LP, 8-APD, 10-Loss of Load
(2) 48 hours

C. (1) 1-High Power, 7-TM/LP, 10-Loss of Load
(2) 24 hours

D. (1) 1-High Power, 7-TM/LP, 10-Loss of Load
(2) 48 hours

Answer: B

Answer Explanation:

A. **Incorrect -** TS 3.3.1 allows 48 hrs to restore

B. **Correct –** T/U's 1,2,7,8,10 affected, and TS 3.3.1 allows 48 hrs to restore

C. **Incorrect -** T/U 2 & 8 are affected and must also be bypassed and TS 3.3.1 allows 48 hrs to restore

D. **Incorrect -** T/U 2 & 8 are affected and must also be bypassed

Question 91 (Q97184)			
Topic:	Evaluate alarms and indications		
Tier/Group:	2/2		
K/A Info:	015 - Nuclear Instrumentation • 2.1 - Conduct of Operations • 2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation		
SRO Importance:	4.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	NO-1-200, Control of Shift Activities Tech Spec 3.3.1		
Comments:	None		

92. 056 - Condensate System

Unit-1 is operating at 100% power when multiple condenser tube failures occur in 12A Waterbox:

- A power reduction was begun.
- At 90% power 12A Waterbox is secured.
- Plant Chemistry determines S/G chemistry is in Action Level 3 for cation conductivity
- It is now 3 hours since the tube failures, power is at 40%. S/G chemistry is in Action level 2 and improving
- Action Level 1 is expected in 3 hours at the current cleanup rate

(1) Predict the impact if no actions are taken for condenser tube failure and;
(2) What action is the appropriate response from this point?

A. (1) Potential breach of 1 fission product barrier;
(2) Place the unit in Mode 2 as quickly as safe operation permits.

B. (1) Potential breach of 1 fission product barrier;
(2) Reduce power to less than 30% within the next 5 hours.

C. (1) Potential breach of 2 fission product barriers;
(2) Terminate the power reduction when Action Level 1 is reached.

D. (1) Potential breach of 2 fission product barriers;
(2) Reduce power to less than 30% then when Action Level 1 is reached, return to 100% power.

Answer: A

Answer Explanation:

A. **Correct** – Condenser tube leak can cause a S/G tube leak if no action taken. AOP-10 states: IF Plant Chemistry determines SG Chemistry is in Action Level 3, as a result of a Condenser tube leak, THEN commence an orderly plant shutdown to be less than 5% power as quickly as safe operation permits PER OP-3 and OP-4. A note preceding this step in AOP-10 states: If SG chemistry levels are reduced below the Action Level 3 value, before or during the power reduction, power level is still required to be reduced below 5%.

B. **Incorrect** - Condenser tube leak can cause a S/G tube leak if no action taken. This action is correct if Action Level 2 was declared at the time of the tube failure. AOP-10 states: Within 8 hours of initiating Action Level 2, reduce power to less than 30% PER OP-3, NORMAL POWER OPERATION.

C. **Incorrect** - Condenser tube leak can cause a S/G tube leak if no action taken, this only affects 1 fission product barrier. 2 barriers is plausible if student thinks ADV operation bypasses containment barrier or abnormal chemistry could enter RCS affecting cladding barrier. This action would be appropriate if the Unit had peaked out in an Action Level 2 condition and Action Level 1 values have been attained. Entry into Action Level 3 requires reducing power to less than 5%.

D. **Incorrect** - Condenser tube leak can cause a S/G tube leak if no action taken, this only affects 1 fission product barrier. 2 barriers is plausible if student thinks ADV operation bypasses containment barrier or abnormal chemistry could enter RCS affecting cladding barrier. This action would be appropriate if the Unit had peaked out in an Action Level 2 condition. Entry into Action Level 3 requires reducing power to less than 5%.

Question 92 (Q41948)			
Topic:	Condenser Tube Leak		
Tier/Group:	2/2		
K/A Info:	056 – Condensate System • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: • A2.05 – Condenser Tube Leakage		
SRO Importance:	2.5*		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	AOP-10 Abnormal Secondary Chemistry		
Comments:	None		

93. 072 - Area Radiation Monitoring

U-1 containment pressure is -0.4 psig and preparing for a negative pressure vent. Containment High Range Radiation Monitor, 1- RI-5317A, reading is erratic due to a power supply problem.

Which **ONE** of the following describes (1) the impact / mitigating strategy of removing this rad monitor from service, and (2) Tech Spec implications of this failure (if applicable)?

A. (1) Negative pressure vent prevented
(2) No TS implications

B. (1) Negative pressure vent prevented
(2) 30 days to return to service per TS

C. (1) Negative pressure vent allowed after verifying RI-5317B in service
(2) No TS implications

D. (1) Negative pressure vent allowed after verifying RI-5317B in service
(2) 30 days to return to service per TS

Answer: D

Answer Explanation:

A. **Incorrect - Negative pressure vent Initial Conditions require only one Containment High Range Rad Monitor, but TS 3.3.10 requires two Containment High Range Rad Monitor with 30 days to return**

B. Incorrect - Negative pressure vent Initial Conditions require only one Containment High Range Rad Monitor, but TS 3.3.10 requires two Containment High Range Rad Monitor with 30 days to return

C. **Incorrect** - Negative pressure vent Initial Conditions require only one Containment High Range Rad Monitor, but **TS 3.3.10 requires two Containment High Range Rad Monitor with 30 days to return**

D. **Correct** - Negative pressure vent Initial Conditions require only one Containment High Range Rad Monitor, but TS 3.3.10 requires two Containment High Range Rad Monitor with 30 days to return

Question 93 (Q97187)			
Topic:	Containment Hi-Range RMS OOS		
Tier/Group:	2/2		
K/A Info:	072 - Area Radiation Monitoring • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the ARM; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations: • A2.01 - Erratic or failed power supply		
SRO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	OI-41B Hydrogen Purge Tech Spec 3.3.10		
Comments:	None		

94. Conduct of Operations

Unit 1 is in Mode 6 with movement of irradiated fuel in progress. A safety tagger requests the Fuel Handling Supervisor (FHS) to independently verify a safety tagged component located on the 69' containment elevation.

Which **ONE** of the following describes the restrictions, if any, placed on the FHS performing the independent verification (IV)?

A. The FHS may not perform the IV under any circumstances.

B. The FHS may perform the IV at any time during the movement of irradiated fuel.

C. The FHS may perform the IV only during suspension of movement of irradiated fuel.

D. The FHS may perform the IV only if the Refueling Machine Operator is a licensed reactor operator

Answer: C

Answer Explanation:

The FHS may leave the refueling machine if a licensed operator is the RFM operator. The FHS may tour the 69' area of containment. The FHS supervises refueling operations with no concurrent duties.

A. **Incorrect** - The designated FHS may perform the independent verification IF fuel handling activities have been suspended.

B. **Incorrect** - NO-1-200, Step 5.1.B.a. states that during fuel loading, transferring or handling an SRO shall be designated the FHS and further states: The FHS shall directly supervise from the Containment 69 foot elevation and have no other concurrent duties. Therefore, the IV cannot be performed by the FHS unless fuel handling activities are suspended.

C. **Correct** - NO-1-200, Step 5.1.B.a. states that during fuel loading, transferring or handling an SRO shall be designated the FHS and further states: The FHS shall directly supervise from the Containment 69 foot elevation and have no other concurrent duties. Therefore, the IV cannot be performed by the FHS unless fuel handling activities are suspended.

D. **Incorrect** - Having a licensed operator on the RFM allows the FHS to leave the RFM as long as he remains on the 69' elevation of the Containment and focused solely on the fuel handling activities. IV cannot be performed by the FHS unless fuel handling activities are suspended.

Question 94 (Q44770)			
Topic:	Determine the limits of travel for the FHS		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.1 - Conduct of Operations • 2.1.40 - Knowledge of refueling administrative requirements.		
SRO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	No previous NRC Exam use		
10 CFR Part 55 Content:	55.43(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:			
Technical references:	NO-1-200, Control of Shift Activities (Step 5.1.B.1.) CNG-OP-1.-1-1000, Conduct of Operations (Step 4.8)		
Comments:	None		

95. 2.1 - Conduct of Operations

Given the following:

- BOTH Units are in Mode 1.
- The shift is manned to the Technical Specification minimum composition.
- The shift has 6 hours remaining.
- An RO has become ill and must leave the site for emergency medical treatment.

Which **ONE** of the following describes the requirements regarding the shift composition and the **MINIMUM** required action in this situation?

A. The RO may leave the site immediately after turnover of responsibilities to the CRO; No replacement required.

B. The RO may leave the site immediately after turnover of responsibilities to the CRS; A replacement **must** arrive within 1 hour.

C. The RO may leave the site immediately after turnover of responsibilities to the CRO; A replacement **must** arrive within 2 hours.

D. The RO may leave the site immediately after turnover of responsibilities to the CRS. A replacement **must** arrive within 6 hours.

Answer: C

Answer Explanation:

Tech Spec 5.2.2.c specifies: "Shift **crew composition may be less than the minimum** requirement of 10 CFR 50.54(m)(2)(i), 5.2.2.a, and 5.2.2.g **for a period of time not to exceed 2 hours** in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements."

A. **Incorrect** - The RO and CRO are licensed to the same level. It may be thought the CRO can fulfill the roles of both the RO and the CRO for the remainder of shift.

B. **Incorrect** - Many of the Tech Specs and Tech Spec Surveillance requirements have completion times/Frequencies of 1 hour making this a plausible distractor.

C. **Correct** - **2 hours** is the correct answer.

A. **Incorrect** - Many of the Tech Specs have completion times of 6 hours making this a plausible distractor.

Question 95 (Q54961)			
Topic:	Administrative requirements for minimum shift staffing		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.1 - Conduct of Operations • 2.1.5 - Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.		
SRO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	Technical Specification 5.2.2.c for Unit Staff		
Comments:	None		

96. 2.2 - Equipment Control

Given the following:

- Unit-2 is operating at 100% power
- Following completion of shift turnover, at 1830 on a Saturday evening, the 2B DG is declared out of service due to problems with the fuel oil transfer pump.
- You have directed the 2B DG be removed from service.

Which **ONE** of the following offsite organizations must you, as the CRS or Shift Manager, notify per OI-21B, 2B DIESEL GENERATOR?

A. The responsible Work Week Manager
B. The Nuclear Regulatory Commission (NRC)
C. Southern Maryland Electric Co-operative (SMECO)
D. System Operator - Transmission Systems Operation (SO-TSO)

Answer: D

Answer Explanation:

A. The Work Week Manager must be notified to assist the OWC in completing an updated risk assessment, however, this notification is not required by OI-21B
B. The NRC is not required to be notified of DG unavailability per OI-21B. The Resident Inspector would be notified by voice-mail or page per CNG- OP-1.01-2001, Communications and Briefings
C. SMECO is not required to be notified of DG unavailability.
D. OI-21B states: The CRS or SM will contact the SO-TSO when removing a D/G from service during the weekend.

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Question 96 (Q74876)			
Topic:	Reporting requirement for DG OOS		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.2 - Equipment Control • 2.2.17 - Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.		
SRO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(1)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	OI-21B, 2B DIESEL GENERATOR		
Comments:	None		

97. 2.2 - Equipment Control

Given the following conditions:

- Unit-2 is in Mode 5
- 21 AFW Pump is out of service for repairs that will take 2 to 3 days to complete

Which **ONE** of the following describes the **HIGHEST OPERATIONAL MODE** that may be entered with 21 AFW Pump out of service?

A. Mode 5

B. Mode 4

C. Mode 3

D. Mode 2

Answer: B

Answer Explanation:

A. **Incorrect** - Turbine Driven AFW Pumps are required operable in Modes 1, 2 & 3. Plausible if student believes no mode changes allowed due to misunderstanding of LCO 3.0.4

B. **Correct** - LCO 3.7.3.a is the applicable TS and contains a note stating LCO 3.0.4.b is not applicable

C. **Incorrect** - Tech Specs provide an allowance for proceeding to Mode 1/2/3 prior to performing Turbine Driven AFW Pump Surveillance Testing

D. **Incorrect** - Tech Specs provide an allowance for proceeding to Mode 1/2/3 prior to performing Turbine Driven AFW Pump Surveillance Testing

Question 97 (Q51274)			
Topic:	Evaluate a Mode change with a TDAFW Pp OOS		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.2 - Equipment Control • 2.2.35 - Ability to determine Technical Specification Mode of Operation		
SRO Importance:	4.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	Technical Specifications Technical Specification Bases		
Comments:	None		

98. 2.4 - Emergency Procedures / Plan

Using provided reference(s):

During AFW testing it was discovered that there is blockage in the U-2 AFW suction line upstream of the level indicator in 27 Ft. Purge Supply Fan Room. An Auxiliary Operator has been sent for "for cause" fitness-for-duty testing for an unrelated event. Condition Reports have been written.

Which **ONE** of the following is the soonest report required?

A. Four (4) hour Emergency Notification System (ENS) Report

B. Eight (8) hour Emergency Notification System (ENS) Report

C. Twenty-four (24) hour Emergency Notification System (ENS) Report

D. Sixty (60) day Licensee Event Report (LER)

Answer: B

Answer Explanation:

A. **Incorrect** - A 4 hour ENS report, per 10 CFR 50.72(b)(2)(i), is not required since T.S. 3.7.3 Action F states "LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status".

B. **Correct** - An 8 hour report ENS is required, under 10 CFR 50.72(b)(3)(v), due to the blockage affecting the ability to removal residual heat.

C. **Incorrect** - A 24 hour ENS report **may** be determined necessary per 10 CFR 26.719(b)(2) when results of fitness for duty screening are obtained. However, this is not the soonest report required.

D. **Incorrect** - A 60 day LER may be determined necessary per either 10 CFR 50.73(a)(2)(i)(B) or 50.73(a)(2)(vii)(B). However, this is not the soonest report required.

Question 98 (Q50852)			
Topic:	AFW System Impairment reporting requirements		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.4 - Emergency Procedures / Plan • 2.4.30 - Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.		
SRO Importance:	3.8		
Proposed references to be provided to applicant:	CNG-NL-1.01-1004, Regulatory Reporting Att. 2 Tech Spec 3.7.3, Auxiliary Feedwater (AFW) System Tech Spec Basis 3.7.3 AFW System		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(1)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	CNG-NL-1.01-1004, Regulatory Reporting Tech Spec 3.7.3, Auxiliary Feedwater (AFW) System		
Comments:	None		

99. 2.3 - Radiation Control

U-1 is in Mode 5 with the following conditions:

- RCS pressure 150 PSIA
- PZR level slowly lowering
- Component Cooling RMS in alarm
- Component Cooling Head Tank level slowly rising

Which **ONE** of the following choices identifies the correct AOP to enter and the **EARLIER** action to take in that procedure?

A. AOP-2A Excessive Reactor Coolant Leakage;
Initiate HPSI flow with 11 & 13 HPSI pumps

B. AOP-2A Excessive Reactor Coolant Leakage;
Verify all available charging pumps operating

C. AOP-3B Abnormal Shutdown Cooling Conditions;
Start Charging pumps as necessary to maintain RCS level

D. AOP-3B Abnormal Shutdown Cooling Conditions;
Shut Component Cooling containment isolation valves (1-CC-3832 & 1-CC-3833)

Answer: C

Answer Explanation

A. **Incorrect** - AOP-2A is incorrect procedure but there is a section for RCS leakage (shutdown) but only for Mode 3, not Mode 5 and initiating HPSI flow is in this section

B. **Incorrect** - AOP-2A is incorrect procedure but there is a section for RCS leakage shutdown but only for Mode 3, not Mode 5 and verifying all charging pumps running is in this section

C. **Correct** - AOP-3B is the correct procedure and starting charging pumps is step V.E.1

D. **Incorrect** - AOP-3B is the correct procedure but isolating containment is step V.F.10.1

Question 99 (Q97193)			
Topic:	CC RMS		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.3 - Radiation Control • 2.3.15 - Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.		
SRO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(4)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A new question		
Technical references:	AOP-2A Excessive Reactor Coolant Leakage AOP-3B Abnormal Shutdown Cooling Conditions		
Comments:	None		

100. Use of the "Follow-up Communications" form

Using Provided References:

Given the following conditions:

- 0900 - A Seismic Event, resulting in a Loss of Offsite Power occurs
- 0912 - The Shift Manager declares an Alert, due to the seismic event
- 0925 - An aftershock occurs resulting in 14 4KV Bus deenergizing
- Investigation, by the Electricians, indicates 14 4KV Bus is grounded. Electricians believe it will take several hours to clear the ground so the Bus can be re-energized.

Which **ONE** of the following is the appropriate form to complete for notifying offsite agencies of events occurring since the Alert declaration and why?

A. EP-Form-ALL23R0 , "Initial Notification Form", since this is a significant plant condition change resulting in Alert criteria being met for a different EAL.

B. EP-Form-ALL23R0, "Initial Notification Form", since conditions warrant an upgrade to an EAL at a higher classification.

C. EP-Form-ALL22R0, "Follow-up Communications Form", since this is a significant plant condition change resulting in Alert criteria being met for a different EAL

D. EP-Form-ALL22R0, "Follow-up Communications Form", since conditions warrant an upgrade to an EAL at a higher classification.

Answer: C

Answer Explanation:

A. **Incorrect** - Although this is a significant plant change, EAL SA1.1 is NOT an upgrade in classification from HA1.1. Initial Notification is NOT used to inform offsite agencies.

B. **Incorrect** - EAL SA1.1 is NOT an upgrade in classification from HA1.1 and Initial Notification is NOT used to inform offsite agencies.

C. **Correct** - EAL SA1.1 is NOT an upgrade in classification from HA1.1. Follow Up Communication is used to inform offsite agencies of this change in plant status.

D. **Incorrect** - This is a significant plant change, EAL SA1.1 is NOT an upgrade in classification from HA1.1. If an upgrade in EAL occurred, Initial Notification is used to inform offsite agencies.

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Question 100 (Q97013)			
Topic:	Use of the "Follow-up Communications" form		
Tier/Group:	Generic Knowledge & Abilities		
K/A Info:	2.4 - Emergency Procedures / Plan • 2.4.40 - Knowledge of SRO responsibilities in emergency plan implementation		
SRO Importance:	4.5		
Proposed references to be provided to applicant:	EAL Hot Chart		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Technical references:	ERPIP 3.0 EAL Mode 1-4 Chart		
Comments:	None		