Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.0	

1 K/A # 000007 CE/E02 EK1.3 Knowledge of the operational implications of annunciators and conditions indicating signals, and remedial actions associated with the Reactor Trip Recovery.

Proposed Question:

Given the following conditions:

- The plant was tripped from 100% power due to loss of load.
- The operating crew has completed EOP-1.0, "Standard Post-Trip Actions", and transitioned to EOP-2.0, "Reactor Trip Recovery".
- Two minutes later, a forklift operator delivering barrels of EHC fluid to the turbine building loses control of the forklift and tears a large hole in the bottom of Condensate Storage Tank T-2.
- Alarm EK-1115 CONDENSATE STORAGE TANK T-2 LO-LO Level is received.
- The operating crew secures all AFW pumps to prevent damage after receiving a low suction pressure trip on AFW P-8A.
- Both Main Feedwater Pumps are operating at minimum speed.

Which of the following describes the actions the crew is directed to take to restore Steam Generator feedwater, per EOP-2.0?

A.) Maintain PCPs operating and align service water to P-8C.
B.) Secure PCPs and use the Main Feed Reg Valve Bypasses.
C.) Secure PCPs and align service water to P-8C.
D.) Maintain PCPs operating and use the Main Feed Reg Valve Bypasses.

Proposed Answer: D

Explanation:

A) Incorrect - EOP-2 does not allow the use of service water. The listed action is from EOP-7, "Loss of All Feedwater Recovery", which is not appropriate for these conditions.
B) Incorrect - EOP-2 does not direct securing primary coolant pumps for a loss of feedwater. The listed action is from EOP-7 which is not appropriate for these conditions.
C) Incorrect - EOP-2 does not direct securing primary coolant pumps for a loss of feedwater, this is from EOP-7 which is not appropriate for these conditions.
D) **Correct** - EOP-2 allows the operator to recover steam generator levels using Main Feedwater. Staying in EOP-2 is preferred to transitioning to EOP-7 since it allows the plant to maintain PCPs that will aid in a more controlled shutdown.

Technical Reference(s): EOP-2 step 8, page 5, and associated basis.

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

Question Source:	Bank #	☐	
	Modified Bank #	☐	
	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☒	3.0
	Comprehension or Analysis	☐	
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments:
TBAB_T06.00

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	2.5	

2 K/A # 000008 G 2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized as they apply to pressurizer vapor space accident.

Proposed Question:

During a routine containment tour on 'B' shift with the plant at full power the upper sensing line for a pressurizer level indication shears off causing a vapor space LOCA. The HP Tech that was accompanying the AO on rounds calls from the airlock and says that there was a loud bang followed by a squealing noise. He thinks the AO has had a heart attack and says he needs help to get the AO out of containment. Dose rates in the general area have risen to 40 rem/hr and the rescue is expected to take 10 minutes.

To properly accomplish the rescue, the Shift Manager will be required to ...

A.) Tell the HP tech to find someone and remove the AO from containment.
B.) Assign two people to the rescue and have them briefed by HP supervisor prior to entry.
C.) Assign two people to the rescue, have them briefed by HP supervisor, and get Site VP approval.
D.) Find two volunteers, have them briefed by HP supervisor, and provide lapel air sampler.

Proposed Answer: B

Explanation:
A) Incorrect - Individuals that are allowed to exceed the normal limits must be briefed prior to any rescue efforts.
B) **Correct** - Individuals that are assigned a task that will exceed normal limits (5 rem) must be briefed prior to any rescue efforts.
C) Incorrect - There is no requirement to get the Site VPs approval.
D) Incorrect - There is no requirement to find volunteers or get duty HP approval for the given conditions. These apply only to exceeding 25 rem.

Technical Reference(s): EI 2.1, "Site Emergency Director", page 4 and attachment 1

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

Question Source: Bank # ☐
Modified Bank # ☐

	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5
10 CFR Part 55 Content:	55.41	b.12	
	55.43		

Comments:
GAT objective

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.4	
3	K/A # 000009 EA2.13 Ability to determine or interpret charging pump flow indication as it applies to a small break LOCA.		

Proposed Question:

With the plant at normal operating pressure and Charging Pump P-55A in service, the controlled bleedoff line for Primary Coolant Pump P-50A breaks off between the pump and the first manual isolation valve.

What is the response of charging system flow to a PCS leak in this location?

A.) Raise by 0 to 1 gal/min
B.) Raise by 2 to 3 gal/min
C.) Raise by 4 to 8 gal/min
D.) Lower by 1 to 2 gal/min

Proposed Answer: A

Explanation:
A) **Correct** - This LOCA will result in a reduction in flow to the VCT but would not significantly change charging flow, The leak rate through the PCP pump will be slightly higher since it now discharges to containment atmosphere and not to the VCT that is slightly pressurized.
B) Incorrect - This is approximately the leak rate created by this failure.
C) Incorrect - This is approximately the leak rate if there were not check valves preventing the controlled bleedoff from the other pumps to reach the break.
D) Incorrect - This answer might be chosen by a candidate who believed this leak was down stream of the stop check valve.

Technical Reference(s): M-202 sheet 1 and M-209 sheet 1

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☒ 4.0

10 CFR Part 55 Content: 55.41 b.5
55.43

Comments: TBCORE_CK05.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.5	
4	K/A # 000011 EA1.17 Ability to operate and monitor the safety parameter display system as it applies to a Large Break LOCA.		

Proposed Question:

During a Large Break Loss of Coolant Accident inside containment the operator notes that the Plant Process Computer (PPC) displayed value for containment pressure has changed color from MAGENTA to WHITE.

How is this information verified on the PPC and what is its significance?

A.) Depress "URGNT" hardkey. Containment pressure is now LESS THAN the alarm level setpoint.
B.) Depress "ALARM" hardkey. Containment pressure is now ABOVE the alarm level setpoint.
C.) Depress "EVENT" hardkey. A Containment High Pressure (CHP) has just actuated.
D.) Depress "UPDATE" hardkey. Criteria for resetting Containment High Pressure are now met.

Proposed Answer: A

Explanation:

a. **CORRECT** - Per PPC operating manual the URGNT hardkey is used to access the information, and it is interpreted as shown here.
b. Incorrect - Though this hardkey does exist, it would not be used for the condition; candidate also misinterprets the significance of the indication.
c. Incorrect usage of this hardkey; candidate also misinterprets the significance of the indication.
d. Incorrect - UPDATE key is a commonly used key on the PPC, but not for this application; further, candidate misinterprets significance of the indication.

Technical Reference(s): PPC User's Manual, pages 5-1 and 5-11.

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

Question Source: Bank # ☒
Modified Bank # ☐

	New	☐	
Question History:	Last NRC Exam	2003	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.0
	Comprehension or Analysis	☐	
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments: TBAA_E02.03

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.0	

5 K/A # 000015/17 AK3.02 Knowledge of the reasons for the CCW lineup and flow paths to the RCP oil coolers as they apply to the Reactor coolant Pump Malfunctions (Loss of RC Flow)

Proposed Question:

Which of the following describes the Component Cooling Water interlock associated with Primary Coolant Pump P-50A?

a. Automatically trips the PCP if CCW FLOW to the PCP integral heat exchanger drops to less than 80 gpm, to protect PCP seals from overheating and damage.

b. Automatically trips the PCP if CCW TEMPERATURE out of the integral heat exchanger exceeds 175º F to protect from overheating the thrust bearing.

c. Prevents starting the PCP if CCW FLOW to the integral heat exchanger is less than 80 gpm to ensure adequate oil cooling capability.

d. Prevents starting the PCP if CCW PRESSURE supplied to the integral heat exchanger is less than 80 psig to ensure adequate oil cooling capability.

Proposed Answer: C

Explanation:
A) Incorrect, there is no pump trip interlock for the PCP on CCW Flow.
B) Incorrect, there is no pump trip interlock for the PCP on CCW Temperature
C) Correct, there is a start permissive associated with CCW flow to the primary coolant pump to ensure that it is adequately cooled prior to being run to avoid damage/failure.
D) Incorrect, there is no start permissive associated with CCW pressure for the primary coolant pump.

Technical Reference(s): ARP-5, window 31

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam ______________

Question Cognitive Level: Memory or Fundamental Knowledge ☒ 3.5
Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 b.5
55.43 ______

Comments: CCW_CK10.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	2.7	
6	K/A # 000022 AK1.02 Knowledge of the operational implications of the relationship of charging flow to pressure differential between charging and RCS as it applies to Loss of Reactor Coolant Pump Makeup.		

Proposed Question:

Given;

- Charging Pump P-55A is out of service for maintenance.
- Charging Pumps P-55B and P-55C are in manual, with P-55B in-service.
- A loss of all Forced Circulation has occurred
- PCS pressure has returned to 2060 psia

How will indicated charging flow change, and why? Indicated flow will...

A.) lower because of reduced PCP controlled bleedoff.
B.) not change due to design of Palisades charging pumps.
C.) rise because Tave will lower as a result of the loss of pump heat.
D.) lower because core delta T will rise causing coolant volume to swell.

Proposed Answer: B

Explanation:
A) Incorrect - Candidate believes that controlled bleedoff flow will affect charging. With a constant speed charging pump in service this will have no effect.
B) **Correct** - B Charging Pump is a constant speed, positive displacement pump.
C) Incorrect - Reflects the effect on PCS volume the loss of pump heat may have.
D) Incorrect - Reflects the effect on PCS volume the larger delta T that will be required will cause.

Technical Reference(s): GFES reference, CVCS Lesson Plan

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History:	Last NRC Exam	________	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5
10 CFR Part 55 Content:	55.41	b.8	
	55.43	________	

Comments: CVCS_CK02.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.9	
7	K/A # 000025 AK1.01 Knowledge of the operational implications of the loss of RHRS during all modes of operation.		

Proposed Question:

Given the following conditions:

- The PCS is being filled from Reduced Inventory.
- It is day 5 of a forced outage to replace a PCP seal package.
- Current PCS level is 628' 5".
- Both SGs have level at approximately 50%.
- Current Average Qualified CET temperature is 140 ^{o}F.
- Shutdown Cooling has been lost.

The PCS will reach 200 ^{o}F in _______ minutes.

A.) 14 to 18
B.) 20 to 24
C.) 30 to 34
D.) 60 to 70

Proposed Answer: B

Explanation:
a. Plausible if candidate uses incorrect curves or data points. Incorrect curves or data points used.
b. Using ONP-17, Attachment 1, page 4, intersection of 5 day line and 140 ^{o}F initial temperature is approximately 21 minutes.
c. Plausible if candidate uses incorrect curves or data points. Incorrect curves or data points used.
d. Plausible if candidate uses incorrect curves or data points. Incorrect curves or data points used.

Technical Reference(s): ONP-17, Loss of Shutdown Cooling

Proposed references to be provided to the applicants during examination: ONP-17 curves

Question Source: Bank # X

	Modified Bank #	☐	
	New	☐	
Question History:	Last NRC Exam	2000	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5
10 CFR Part 55 Content:	55.41	b.8	
	55.43		

Comments: SDC-CK11.0b

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.4	
8	K/A # 000026 G2.4.11 Knowledge of abnormal condition procedures as they apply to Loss of Component Cooling Water.		

Proposed Question:

The plant is operating at full power. An auxiliary operator (AO) is in the field performing an oil change on the motor of Component Cooling Water Pump P-52B. The AO calls the control room to report that there is a lot of noise coming from the pump seal area of the operating CCW pump, P-52A. While discussing what actions to take in the control room, the operating crew receives the following alarms:

- EK-1167 COMPONENT CLG PUMPS P-52A, P-52B, P-52C TRIP
- EK-1169 COMPONENT CLG PUMP DISCHARGE LO PRESS

The control room enters ONP-6.2, Loss of Component Cooling. The AO in the field reports that CCW P-52A has now tripped and that the seal appears to have been damaged causing an approximately 2 gpm leak. CCW Surge T-3 level is 50% and lowering slowly. P-52C is NOT running.

Per ONP-6.2 should the control room operator manually start P-52C, and why?

A.) No. P-52C should have started in standby and cannot be started until the cause of this failure is known.
B.) No. Starting P-52C will cause more leakage, it cannot be started until P-52A is isolated.
C.) Yes, since P-52C should have started in Standby and CCW surge tank level is sufficient and will auto fill.
D.) Yes, since the AO in the room can verify the required valve lineup and make necessary heat exchanger adjustments.

Proposed Answer: C

Explanation:
A) Incorrect - The pump should have started in standby but the procedure does not require that this be resolved prior to starting the pump.
B) Incorrect - Starting P-52C will probably cause more leakage but there is no requirement in the procedure to stop all leakage. This leakage will be compensated for by the make up to the CCW Surge Tank. The AO is in the room and would be able to isolate the pump shortly after the P-52C was started. A crew may choose to wait until the pump is isolated, but the procedure does not require it, and waiting too long could preclude the

ability to attempt to start P-52C. (10 minute limit) This would result in a plant trip. This is a much higher consequence than additional leakage.

C) **Correct -** as stated in step ONP-6.2, Loss of CCW, step 4.1.

D) Incorrect - ONP-6.2 does not require that the valve lineup be verified or that heat exchanger dP be reviewed. While these checks may be performed by the crew after restoration of the pump they are not required as part of ONP-6.2 which is what the question asks.

Technical Reference(s): ONP-6.2, ARP 7

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

Question Source:	Bank #	☐
	Modified Bank #	☐
	New	☒

Question History: Last NRC Exam ______

Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.0

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments: IOTF_CK07.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.6	
9	K/A #	000027 AA2.16 Ability to determine and interpret the actions to be taken if PZR pressure instrument fails low.	

Proposed Question:

With the plant at full power the input signal to the B channel Pressurizer Pressure Control fails low. The B channel of Pressurizer Pressure Control is the in-service channel. Which of the following is an expected procedural direction from the CRS that would restore Pressurizer pressure to normal?

A.) Operate 1/LIC-0101 Heater Control Selector switch to "Channel A" position.
B.) Manually open the pressurizer spray valves using the handswitches.
C.) Take manual control of the in-service pressure controller.
D.) Manually close the pressurizer spray valves using the handswitches.

Proposed Answer: C

Explanation:

A) Incorrect - While the ONP does allow the CRS to direct transfer to the other controller, simply transferring control will not result in pressure returning to normal. Since the out of service controller will be at 50% output signal, pressure will continue to rise. The CRS would also have to provide direction to take manual control or transfer to automatic control.

B) Incorrect - The spray valves cannot be manually opened using the handswitches. There is direction in the ONP to close the sprays using the handswitches if pressure is lowering, that is why this was chosen as a distractor.

C) **Correct** - one of the options provided by the procedure.

D) Incorrect - This action can be taken but it will not aid in recovering pressurizer pressure.

Technical Reference(s): ONP-18, SOP-1A

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

Question Source:

- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.0
	Comprehension or Analysis	☐	

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments: IOTF_CK07.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	4.1	
10	K/A #	000029 EA1.15 Ability to operate and monitor the AFW system as it applies to the ATWS.	

Proposed Question:

With the Plant at full power the control room receives EK-0548, "125V DC BUS UNDERVOLTAGE/TROUBLE" along with several other alarms. Following a quick scan of the panels the Reactor Operators report position indication for CV-0510, S/G E-50A MSIV has been lost and CV-2009, Cont Letdown Isolation has failed closed.

Which of the following is an additional condition that the operators should be able to observe?

A.) CV-1359, Non-Critical Service Water Isolation failed closed.
B.) Auxiliary Feedwater P-8B in-service.
C.) Control Power to Bus 1C lost.
D.) Control Power to Bus 1D lost.

Proposed Answer: B

Explanation:

A. Incorrect - Caused by a failure of another part of the DC system, not by failure of D-11-1.
B. **Correct** - The automatic start of P-8B on a loss of D-11-1 is part of the ATWS modification. This question tests whether or not the student can recognize the a loss of D-11-1 and then if they know that this causes an auto start of P-8B.
C. Incorrect - Failure that is caused by a loss of a different section of the DC system.
D. Incorrect - Failure that is caused by a loss of a different section of the DC system.

Technical Reference(s): ONP-2.3, DBD-1.03

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

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam ____

Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	4.0

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments: AFW_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.9	

11 K/A # 000038 EK3.04 Knowledge of the reasons for the automatic actions provided by each PRM as they apply to the SGTR.

Proposed Question:

During a Steam Generator Tube Rupture which of the following is an automatic action initiated by RIA-0707 Steam Generator Blowdown monitor and the basis for the action?

The monitor signals the ......

A.) isolation of both outer bottom blowdown valves (CV-0770, 0771) to limit the spread of contamination in the blowdown system.
B.) isolation of both surface blowdown valves (CV-0738, 0739) to prevent contamination from reaching chemistry's sample panel.
C.) isolation of PCV-6003, Flash Tank T-29A Pressure Control to minimize Main Condenser contamination.
D.) trip of both S/G blowdown pumps to prevent spread of contamination from the blowdown system.

Proposed Answer: A

Explanation:
A.) **Correct** - The outer isolation valves get a close signal and isolate the blowdown system.
B.) Incorrect - Chemistry's continuous sample points come off the bottom blowdown lines and upstream of RIA-0707
C.) Incorrect - This valve doesn't receive a signal from RIA-0707.
D.) Incorrect - RIA-0707 doesn't provide a signal to the blowdown pumps.

Technical Reference(s): ARP-8, and P&ID M-223 Sheet 1

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

Question Source:		
	Bank #	☐
	Modified Bank #	☐
	New	☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☒	3.0
	Comprehension or Analysis	☐	

10 CFR Part 55 Content:	55.41	b.5
	55.43	

Comments: RMS_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.3	

12 K/A # 000040 CE/E05 EK2.1 Knowledge of the interrelations between the Excess Steam Demand and the components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question:

With the plant at full power the main steam line on the B Steam Generator fails inside containment. Which of the following instruments are affected by degraded containment conditions in a way that prevents using direct readings in determining if Safety Injection Throttling Criteria are met?

A.) Pressurizer Level
B.) Steam Generator Level
C.) Reactor Vessel Level Monitoring
D.) PCS Temperature

Proposed Answer: A

Explanation:

A.) Correct - Pressurizer level must be corrected for containment conditions before it is used to verify SIAS throttling conditions.

B.) Incorrect - Steam generator level does have to be corrected to read level accurately, however only a rising trend is required to meet throttling criteria and this can be done without correction.

C.) Incorrect - This parameter is used for validating throttling criteria but is not affected by the given conditions.

D.) Incorrect - This parameter is used for throttling criteria but is not affected by containment conditions, however during an ESDE the loss of PCP often requires the use of CETs, but this does not match the stem of the question.

Technical Reference(s): EOP-4 Step 25 Throttling criteria and EOP supplements 9 and 11

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

Question Source:

- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History:	Last NRC Exam	________	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5
10 CFR Part 55 Content:	55.41	b.7	
	55.43	________	

Comments: TBAD_TBCORE_CK05.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.2	

13 K/A # 000054 CE/E06 EK1.02 Knowledge of the operational implications of the components, capacity, and functions of emergency systems as they apply to the Loss of Feedwater.

Proposed Question:

The Plant has tripped due to a Loss of Off-site power at 1300. Emergency D/G 1-1 will not start. Power control has been contacted and is unable to project when power will be restored to the switchyard.

Assuming operators take all required actions which of the following design capacities will be the FIRST to be exceeded before power is restored?

A.) Diesel Generator Fuel oil inventory.
B.) Condensate inventory.
C.) Station Battery Voltage.
D.) ADV Nitrogen backup supply.

Proposed Answer: B

Explanation:
A.) Incorrect - The day tank contains 15 hours of fuel oil and there are no given problems with fuel oil transfer pumps.
B.) **Correct** - The required inventory is sufficient to supply the AFW system for 8 hours. Without power there is no way to retrieve addition volume from other tanks.
C.) Incorrect - The station batteries are designed for only 4 hours, however the procedures direct supplying both busses from a single operating diesel generator.
D.) Incorrect - Operating air for the ADVs can be provided via a D/G 1-2 powered air compressor. The back-up Nitrogen supply is from the bulk nitrogen tank and not a nitrogen bottle station.

Technical Reference(s): Technical Specification 3.7.6 Basis, page 3.7.6-2, LCO

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

Question Source: Bank # ☐
Modified Bank # ☐

	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.5
10 CFR Part 55 Content:	55.41	b.8	
	55.43		

Comments: AFW_CK13.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.0	
14	K/A # 000055 G 2.1.3 Knowledge of shift turnover practices as they apply to a station blackout.		

Proposed Question:

The state of Michigan has been affected by a large system blackout. Palisades tripped and initially neither diesel generator would supply its associated bus. Subsequently the on-shift crew was able to replace control power fuses on the output breaker for D/G 1-1 and it is now supplying Bus 1C. Shift management has decided that it is the right time for the on-shift crew to be relieved.

Which of the following information is **NOT** required to be reviewed during turnover between the off-going and on-coming NCOs?

A.) All completed steps in the work order to replace fuses on the D/G 1-1 output breaker.
B.) Status of in-progress Station Battery load stripping per EOP Supplements 7 and 8.
C.) The station log and associated notes.
D.) Current control bands and methods of control.

Proposed Answer: A

Explanation:
A.) **Correct** - This would not be an operations evolution and the status of the completed steps is not a required turnover item.
Distractors B, C, and D are all required by the conduct of operation procedures.

Technical Reference(s): FP-OP-COO-01 Rev. 1, attachment 14

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge ☒ 2.0
- Comprehension or Analysis ☐

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments: APOC_T13.00

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	2.7	
15	K/A #	000056 AA2.78 Ability to determine and interpret bus voltmeters as they apply to the loss of offsite power.	

Proposed Question:

The plant is at full power when a problem with a breaker failure relay in the switchyard causes a loss of F bus. After stabilizing the plant the operating crew observes the following indications;

- D/G 1-1 is not operating.
- D/G 1-2 is running with its output breaker closed.
- Bus 1C is indicating 2380 VAC.
- Bus 1D is indicating 2405 VAC.
- Bus 1E is indicating 2380 VAC.
- Bus 12 is indicating 0 Amps.
- Bus 13 is indicating 81 Amps (normal).
- Station Power Transformer 15 is indicating 110 Amps (normal).
- Station Power Transformer 16 is indicating 0 Amps

Based on these indications which of the following statements is accurate?

A.) D/G 1-1 should have started.
B.) Bus 1E should have de-energized.
C.) Bus 12 should be carrying some load.
D.) Transformer 16 should be carrying some load.

Proposed Answer: C

Explanation:
A) Incorrect - All safety related busses will have fast transferred to startup power.
B.) Incorrect - Bus 1E also fast transfers to startup power.
C) **Correct** - Although there was a problem with Bus 1D it is now being supported by its diesel generator and Bus 12 doesn't load shed.
D) Incorrect - There is a lockout on the breaker for transformer 16 (PZR heaters) to ensure that it doesn't re-energize without operator action (plant modification ~ 1 year ago).

Technical Reference(s): E-17 Sheet 10, ONP-18

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

Question Source:	Bank #	☐
	Modified Bank #	☐
	New	☒

Question History: Last NRC Exam ______

Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.5

10 CFR Part 55 Content:	55.41	b.5
	55.43	

Comments: SPS_CK11.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.2	

16

K/A # 000057 AA1.05 Ability to operate and/or monitor backup instrument indications as they apply to the loss of vital AC instrument bus.

Proposed Question:

A loss of Preferred AC bus Y-10 has caused the following indications (along with several others) to fail low. Of the indications listed which one does **NOT** have a redundant instrument that operators could use to trend level changes in the associated components?

A.) LIA-0102A PZR Level (Cold Calibrated)
B.) LIA-0920 CCW Surge Tank Level
C.) LIA-0365 SIT T-82A level
D.) LI-0751A A Steam Generator Level

Proposed Answer: C

Explanation:
A.) Incorrect - This instrument does have a redundant indication.
B.) Incorrect - This instrument does have a redundant indication.
C.) **Correct** - Level switches can provided a form of redundancy, but cannot be used to provide a trend.
D.) Incorrect - This parameter has 3 additional indications (this is a steam generator level input to the RPS and is powered by Y-10).

Technical Reference(s): ONP-24.1

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge ☒ 2.0
- Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 b.7

55.43 ______

Comments: SIS_CK14.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.6	

17 K/A # 000062 AK3.02 Knowledge of the reasons for the automatic actions (alignments) within the nuclear service water system resulting from the actuation of the ESFAS as they apply to a Loss of Nuclear Service Water.

Proposed Question:

When a Safety Injection Actuation signal is generated the Service Water inlet isolation valve CV-0869, for Containment Air Cooler #4 goes closed. Why does this valve close?

A.) To ensure there is adequate Service Water to the other three coolers during a loss of Bus 1D.
B.) To ensure there is adequate Service Water to the other three coolers during a loss of Bus 1C.
C.) To isolate Containment Air Cooler #4 since it is not rated for use in accident conditions.
D.) To minimize the potential for vapor binding of the SW piping inside containment.

Proposed Answer: B

Explanation:
A.) Incorrect - While the statement is true it is not the reason it is closed.
B) **Correct** - This train doesn't rely on SW. It isolates and raises flow to the other coolers.
C.) Incorrect - It is not isolated because it is not rated for accident conditions. In fact the outlet valve still gets an open signal so the CAC is not even isolated in an emergency.
D) Incorrect - Since the downstream valve is open, there is no vapor binding concern.

Technical Reference(s): DBD 1.02 section 3.2.6.3

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☒ 3.0

Comprehension or Analysis ☐

10 CFR Part 55 Content:	55.41	b.4
	55.43	

Comments: SWS_CK24.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Importance Rating	3.7	
18	K/A #	000065 AK3.08 Knowledge of the reasons for the actions contained in the EOP for loss of instrument air as they apply to the loss of instrument air.	

Proposed Question:

The plant has experienced a plant trip followed by a complete loss of instrument air. Per EOP-9, "Functional Recovery Procedure" the CRS directs the NCO to match the handswitch positions of the air operated valves with their failed positions.

Why does the EOP direct this action?

A.) To conserve inventory of nitrogen back up systems.
B.) To ensure that valves do not spontaneously change position while air is lost.
C.) To ensure that the valves do not reposition when air is restored.
D.) To place all valves to their safety related position.

Proposed Answer: C

Explanation:

A.) Incorrect - A valve that was being held open by a nitrogen back up system would not be repositioned. This attachment does not conserve nitrogen back up system inventory.

B.) Incorrect - Moving the handswitches will not affect valves during the event. Since there is no air pressure the handswitches cannot keep valves from changing position.

C.) **Correct** - The handswitches are placed in the failed position to ensure that the valves do not reposition during restoration of air.

D.) Incorrect - There are no directions in the checklist to match valves with their safety related position. The checklist directs matching the handswitch regardless of whether it has failed to its safety position or not.

Technical Reference(s): Supplement 36 and its related basis.

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History:	Last NRC Exam	______	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.5
	Comprehension or Analysis	☐	
10 CFR Part 55 Content:	55.41	b.10	
	55.43	______	

Comments: TBAH_TBCORE_CK01.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Importance Rating	2.5	
19	K/A #	000059 G 2.2.24 Ability to analyze the affect of maintenance activities on LCO status as they apply to accidental liquid radwaste releases.	

Proposed Question:

Given the following conditions:

- The plant is in MODE 5.
- PCS is at 623' with S/G nozzle dams installed.
- The Shutdown Cooling System is controlling PCS temperature at 103°F.
- T-91, Utility Water Storage Tank, batch release is in progress
- The batch release calculation assumed two Service Water Pumps in operation.
- Main Condenser East and West Waterboxes are open for installation of a modification.
- Service Water Pumps P-7A and P-7B are in service.
- Service Water Pump P-7C is tagged out for bearing replacement.
- Both Cooling Tower Pumps P-39A/B are available, but not in service.
- The Canal Sample Pump is in service.

The Secondary AO reports that the basket strainer on P-7A discharge appears to be clogging as indicated by differential pressure trending steadily upward . A crew brief is held to discuss removing P-7A from service to disassemble and clean the basket strainer.

For the above plant conditions, what are the implications of the proposed maintenance on the SW pump basket strainer?

A.) Plant will have Required Actions per LCO 3.4.8 PCS Loops - MODE 5, Loops not filled. Terminate the batch release before stopping P-7A.
B.) Plant will have Required Actions per LCO 3.4.8 PCS Loops - MODE 5, Loops not filled. Establish 15 minute periodic sampling of batch release flow.
C.) Plant will have Required Actions per LCO 3.7.8 Service Water System. Terminate the batch release before stopping P-7A.
D.) Plant will have Required Actions per LCO 3.7.8 Service Water System. Start a Dilution Water Pump before stopping P-7A.

Proposed Answer: A

Explanation:
A.) **Correct** - The Service Water LCO does not apply below MODE 4. The batch release

requires that the dilution flow requirements be met. Securing a Service water pump would violate these requirements and require that the batch be secured while the release calculations are updated. In Mode 5 with the loops not filled two SDC trains are required to be operable. To be operable each train needs to have a separate SW pump. With only one pump operable only one train could be considered operable.

B.) Incorrect - See explanation in A above.
C.) Incorrect - See explanation in A above.
D.) Incorrect - See explanation in A above.

Technical Reference(s): TS LCO 3.4.8 and LCO 3.7.8 and HP-6.4

Proposed references to be provided to the applicants during examination: LCO 3.4.8
LCO 3.7.8

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☒ 3.0

10 CFR Part 55 Content: 55.41 b.10
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Importance Rating	3.6	

20 K/A # 000060 AA2.06 Ability to determine and interpret the valve lineup for release of radioactive gases as they apply to accidental gaseous radwaste release.

Proposed Question:

Planned maintenance is to be conducted on RIA-1113, Waste Gas Discharge Monitor. To conduct the maintenance, the following valve lineup was performed, per SOP-18A, Radioactive Waste System - Gaseous:

- MV-WG117, Waste Gas Radiation Monitor Inlet **CLOSED**
- MV-WG119, Waste Gas Radiation Monitor Outlet **CLOSED**
- MV-WG118, Waste Gas Radiation Monitor Bypass **OPEN**

AFTER the maintenance is completed, the following valve lineup is noted:

- MV-WG117, Waste Gas Radiation Monitor Inlet **OPEN**
- MV-WG119, Waste Gas Radiation Monitor Outlet **OPEN**
- MV-WG118, Waste Gas Radiation Monitor Bypass **OPEN**

Which one of the following correctly evaluates the above AFTER maintenance valve lineup?

A.) The monitor inlet valve should be CLOSED, since the monitor is placed in service only as part of preparations for a planned gas batch release.
B.) The monitor bypass valve is OPEN to allow a release path in case RV-1111, Waste Gas Surge Tank Relief, lifts.
C.) The monitor outlet valve is OPEN to ensure the monitor will not overpressurize when RV-1111, Waste Gas Surge Tank Relief, lifts.
D.) The monitor bypass valve should be CLOSED, to ensure the monitor will sense a high radiation condition.

Proposed Answer: D

Explanation:
A) Incorrect - This monitor is normally aligned for service.
B) Incorrect - The bypass should be closed, this is the reason the valve is opened for maintenance.
C) Incorrect - Candidate misapplies function of monitor's normal flowpath.
D) **Correct** -,This is the required in-service line up, per procedure.

Technical Reference(s):	SOP-18A, 5.1.5	
	M-223, sh. 1A, E-2	M-211, sh. 3, G-4

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

Question Source:	Bank #	☐
	Modified Bank #	☒ RMS_CK16.0-1-1
	New	☐

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5

10 CFR Part 55 Content:	55.41	b.11
	55.43	

Comments: RMS_CP03.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Importance Rating	3.6	

21 K/A # 000061 AA1.01 Ability to operate and/or monitor the automatic actuations as they apply to the area radiation monitoring system alarms.

Proposed Question:

The plant is nearing the end of a long production run with elevated PCS activity due to a failed fuel rod. Operators note a rising sump trend along with elevated counts on the Containment Gas monitor. The crew enters ONP 23.1 for a PCS leak and the reactor is manually tripped. During the immediate actions of EOP-1 the NCO at the controls notes that EK-1363 "CONTAINMENT HI RADIATION" has annunciated.

Based on this alarm which of the following accounts for the above conditions and describes any required actions?

A.) At least one containment radiation monitor has gone into WARNING. Ensure containment is isolated.
B.) At least one containment radiation monitor has gone into ALARM. No manual actions are required.
C.) At least one containment radiation monitor has gone into ALARM. Ensure containment is isolated.
D.) At least two containment radiation monitors have gone into ALARM. No manual actions are required.

Proposed Answer: C

Explanation:
A.) Incorrect - Warning function exists, but does not provide containment isolation function.
B.) Incorrect - This alarm will come in when the first containment monitor goes into alarm. The operator is expected to check the monitors when the alarm comes in and manually initiate containment isolation if the alarm is valid.
C.) **Correct** - This alarm will come in when the first containment monitor goes into alarm. Containment doesn't automatically isolate until at least two Containment Radiation monitors go into alarm. This is a knowledge item from EOP-1 because an operator is expected to check the monitors when the alarm comes in and isolate containment if necessary.
D.) Incorrect - This alarm annunciates when one or more containment radiation monitors have alarmed.

Technical Reference(s): ARP-8 EK-1363

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

Question Source:	Bank #	☐
	Modified Bank #	☐
	New	☒
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒ 3.0
	Comprehension or Analysis	☐
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments: RMS_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Importance Rating	3.3	
22	K/A #	000067 AK3.04 Knowledge of the reasons for the actions contained in EOP for plant fire on site as they apply to a plant fire on site.	

Proposed Question:

A fire in the Electrical Equipment room has resulted in a loss of Bus 1D and D/G 1-2. Bus 1C is still being powered by the Safeguards Transformer. The reactor has been tripped and operators are stabilizing the plant and supporting the fire brigade. Per ONP-25.1 "FIRE WHICH THREATENS SAFETY RELATED EQUIPMENT" the CRS directs an NCO to dispatch the safe shutdown AO to align the alternate power supply to the D/G 1-1 Room ventilation fans.

For the stated conditions, this action is taken to ensure that the vent fans ....

A.) do not spuriously start and over cool the room.
B.) are available to prevent engine damage in the event that D/G 1-1 is needed.
C.) are available to maintain room habitability in the event that D/G 1-1 is needed.
D.) are powered by a safety related power supply that is unaffected by the fire.

Proposed Answer: B

Explanation:
A.) Incorrect - This is not the reason for performing this action. If this were a concern outside air temperature would be part of the prerequisites.
B.) **Correct** - Per the caution in ONP-25.1.
C.) Incorrect - The vent fans are needed for engine operability not operator comfort.
D.) Incorrect - This action actually aligns the fans to a non-safety related power supply.

Technical Reference(s): ONP-25.1 step 15, Attachment 21 and SOP-24 attachment 7.

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☒	3.0
	Comprehension or Analysis	☐	

10 CFR Part 55 Content:	55.41	b.5/10
	55.43	

Comments: TBAM_CK05.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Importance Rating	3.3	
23	K/A #	000068 AK2.07 Knowledge of the interrelations between the control room evacuation and the ED/G.	

Proposed Question:

A fire in the cable spreading room associated with the DC power to the ADVs is in progress. The fire causes the ADVs to open. In order to close the ADVs both DC shunt trip buttons have been pushed. Due to the fire and the loss of indication the control room has been evacuated.

If offsite power becomes unavailable to Bus 1C and Bus 1D, would either D/G auto start?

A.) Yes, both diesels will start.
B.) Yes, only D/G 1-1 will start.
C.) Yes, only D/G 1-2 will start.
D.) No, neither D/G will start.

Proposed Answer: A

Explanation:

A.) **Correct** - Pushing the shunt trip push buttons isolates the batteries from everything except the control power for the diesels and Bus 1C and Bus 1D.

B.) Incorrect - Candidate incorrectly believes this requires manual control of the D/G or use of the RLT switches which would mean that only D/G 1-1 could be started.

C.) Incorrect - Candidate incorrectly believes this requires manual control of the D/G and misapplies understanding of the RLTS, and concludes that only D/G 1-2 will start.

D.) Incorrect - Candidate misunderstands the scope of what part of the system is isolated when using the shunt trip pushbutton feature.

Technical Reference(s): ONP-2.3, ONP-20, ONP 25.1

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments: EDG_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Importance Rating	2.6	
24	K/A #	000069 AK1.01 Knowledge of the operational implications of the effect of pressure on leak rate as they apply to a loss of containment integrity.	

Proposed Question:

The plant has experienced an excess steam demand event caused by the failure of a main steamline inside containment. The pressure in containment caused the tubing associated with the containment sump level indication to fail resulting in a leak of the containment sump to East Safeguards that cannot be isolated. Containment pressure was 40 psia when the leak was discovered and estimated to be 3 gpm.

What will the leak rate be if containment pressure is lowered to 20 psia?
(Assume atmospheric pressure is 15 psia.)

A.) 0.60 gpm
B.) 1.37 gpm
C.) 1.50 gpm
D.) 2.12 gpm

Proposed Answer: B

Explanation:
The leak rate will be proportional to the square root of the differential pressure. The candidate has to remember this fact and recognize the units of the containment pressure are PSIA. This is the unit that would be available in the control room. The correct answer is
3gpm*5psig^0.5/25psig^0.5 = 1.37 gpm. The other answers are the result of either using the wrong units, assuming a straight ratio, or both.

Technical Reference(s): GFES

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.5

10 CFR Part 55 Content:	55.41	b.8
	55.43	

Comments: CTMT_CK13.0

Examination Outline Cross-Reference:	Level		RO	SRO
	Tier #		1	
	Group #		2	
	Importance Rating		3.4	
25	K/A #	000074 G 2.2.22 Knowledge of limiting conditions for operations and safety limits as they apply to inadequate core cooling.		

Proposed Question:

The head has just been re-installed and tensioned. The PCS is filled to 628 feet and is stable at 110ºF. Based on decay heat it will take 60 minutes to reach 200ºF if Shutdown Cooling is secured. No primary coolant pumps have been started yet. To support troubleshooting on CV-3006, SDC HX BYPASS, operations has been requested to secure shutdown cooling.

Can Shutdown Cooling be secured, and if so, how long can it be secured without violating Technical Specifications for Shutdown Cooling or PCS Heatup/Cooldown rates?

A.) SDC cannot be secured under these conditions.
B.) SDC cannot be secured for more than 13 minutes.
C.) SDC cannot be secured for more than 26 minutes.
D.) SDC cannot be secured for more than 60 minutes.

Proposed Answer: B

Explanation:
TS allows SDC to be secured for up to an hour. However for the conditions given the heat up rate is such that 20ºF/hour heatup rate limit will be exceeded in 13.3 minutes.

A) Incorrect - Candidate believes conditions do not allow securing SDC,
B) **Correct** - TS allows SDC to be secured for up to an hour. However for the conditions given the heat up rate is such that 20ºF/hour heatup rate limit will be exceeded in 13.3 minutes. PCS will heat up to 200 F in 60 from references provided.
C.) Incorrect - candidate believes the limit on heatup rate is 40F/hr
D) Incorrect - candidate believes that shutdown cooling can be secured for a full hour.

Technical Reference(s): TS 3.4.8, SOP-3, and ONP-17

Proposed references to be provided to the applicants during examination: First Page of TS 3.4.7 and 3.4.8 and SOP 3 Step 7.3.7

Question Source: Bank # ☐
Modified Bank # ☐

	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.5
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

SDC_E05.01

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Importance Rating	3.2	
26	K/A #	CE/A11 AK2.1 Knowledge of the interrelations between the RCS Overcooling and components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	

Proposed Question:

The reactor trips while the NCO Turbine is out of the control room. As a result neither Main Feed Pump is slowed and both Steam Generators are overfilled. When the NCO turbine arrives in the control room he quickly slows both Main Feed Pumps, closes both Main Feed Reg Valves, and plant conditions begin to recover. The following readings represent the plant conditions at the point at which the primary coolant system begins to recover;

- Pressurizer Pressure - 1432 psia
- SIAS has actuated.
- PCS Thot - 480°F
- PCS Tcold - 478°F
- Actual Pressurizer Level - 32%
- A Steam Generator Level - 89%
- B Steam Generator Level - 88%

Assuming no operator action what would be the status of the Pressurizer heaters when pressurizer level recovers to 40%?

A.) Both banks of heaters would be energized.
B.) Only the heaters powered by Bus 1E would be energized.
C.) Only the heaters powered by Bus 1D would be energized.
D.) None of the heaters would be energized.

Proposed Answer: D

Explanation:
A.) Incorrect - SIAS de-energizes Bus 1E which powers one bank of heaters. Bus 1D de-energizes when low PZR level heater cutout occurs.
B.) Incorrect - SIAS de-energizes Bus 1E.
C.) Incorrect - Bus 1D de-energizes when low PZR level heater cutout occurs.
D) **Correct** - Bus 1D heaters do not reset after low PZR level heater cutout and Bus 1E was

load shed on SIAS.

Technical Reference(s): ONP-18, ARP-4, ARP-8, EOP Supplement 5

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

Question Source:	Bank #	☐
	Modified Bank #	☐
	New	☒
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒ 3.0
10 CFR Part 55 Content:	55.41	b.8
	55.43	

Comments: PPCS_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Importance Rating	3.2	

27 K/A # CE/A16 AK1.3 Knowledge of the operational implications of the annunciators and conditions indicating signals, and remedial action association with the Excess RCS Leakage.

Proposed Question:

While operating at full power the control room receives EK-0734 "CHARGING PUMPS SEAL COOLING LO PRESS", the NCO at the boards notes that Charging Pump P-55B has started. Additionally, the NCO notes that sump level is rising.

Initially the containment sump level rate of rise was 0.05 gpm, the VCT was at 78.0%, and the Pressurizer was at 56.9%. After 10 minutes the rate of rise in the sump is 7.9 gpm (actual rate of rise based on level change, not PPC calculation), the VCT is at 73.7%, and the Pressurizer is at 57.1%. PCS Temperature has remained stable.

Which of the following is the required action?

A.) Trip the reactor.
B.) Perform an emergency downpower per ONP-28.
C.) Continue to gather leakrate data and start a downpower per GOP-8.
D.) Attempt to isolate the leak and prepare for a downpower per GOP-8.

Proposed Answer: A

Explanation:
A.) **Correct** - The leakrate is greater than 10 gpm. This is trip criteria per ONP-23.1.
B.) Incorrect - Candidate incorrectly calculates leak rate, or misapplies guidance in the procedure.
C.) Incorrect - Candidate incorrectly calculates leak rate, or misapplies guidance in the procedure.
D.) Incorrect - Candidate incorrectly calculates leak rate, or misapplies guidance in the procedure.
The stem provides some information concerning the leakrate into the sump. This leakrate would not necessarily be indicative of all leakage. The candidate should expect that sump fill rate would lag actual leakrate due to moisture not immediately condensing in areas that would drain directly to the sump. The additional clarification on rate of containment sump rise is due to the fact that the value calculated on the PPC is a 15 minute average.

Technical Reference(s): ONP-23.1

Proposed references to be provided to the applicants during examination: ONP 23.1 Step 4.7

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☒ 2.5

10 CFR Part 55 Content: 55.41 b.10
55.43

Comments: IOTF_T11.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.7	
28	K/A #	003 K1.08 Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the containment isolation systems.	

Proposed Question:

With the plant at 10% power, RIA-1805, Containment Area Radiation Monitor, suffers a short causing it to fail high. While investigating this issue Preferred AC Bus Y-30 de-energizes. What effect does this have on the plant?

A.) A reactor trip due to closure of the MSIV's.
B.) A turbine trip due to closure of the MSIV's.
C.) Loss of Instrument Air to the Pressurizer Spray Valves.
D.) PCP controlled bleed-off being controlled by a relief valve.

Proposed Answer: D

Explanation:
A.) Incorrect - MSIVs do not close on a CHR. Candidate confuses CHP and CHR. Power level also plausibility of these choices as it leads them down the path of turbine only trips.
B.) Incorrect - MSIVs do not close on a CHR. Candidate confuses CHP and CHR. Power level also plausibility of these choices as it leads them down the path of turbine only trips.
C.) Incorrect - Instrument air does not isolate on a CHR.
D.) **Correct** - This will result in a loss/trip of 2 of 4 Containment Area Radiation Monitors (loss of Y-30 results in loss of power to RIA-1807) which will cause a containment isolation on high radiation (CHR). The MSIV's will not close.

Technical Reference(s): ARP-8, EOP Supplement 6

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☒ 2.5

10 CFR Part 55 Content: 55.41 b.7
55.43

Comments: IOTF_CK02.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.5	
29	K/A # 003 K2.02 Knowledge of bus power supplies to the CCW pumps.		

Proposed Question:

Which of the following provides the correct power supply for each of the Component Cooling Water pumps?

	P-52A	P-52B	P-52C
A.)	Bus 1D	Bus 1D	Bus 1C
B.)	Bus 1D	Bus 1C	Bus 1D
C.)	Bus 1C	Bus 1D	Bus 1C
D.)	Bus 1C	Bus 1C	Bus 1D

Proposed Answer: C

Explanation:

A) Incorrect - Candidate incorrectly recalls the power supply configuration of the containment spray pumps as the Component Cooling Water pumps power supply configuration.

B) Incorrect - Candidate incorrectly recalls the power supply configuration of the Service Water pumps as the power supply configuration of the Component Cooling Water pumps.

C.) **Correct** - This is the correct power supply configuration for the Component Cooling Water pumps.

D) Incorrect - Candidate fails to identify the correct power supplies for the Component Cooling Water pumps.

Technical Reference(s): P&ID E-3 sheet 1

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

Question Source: Bank # ☐
Modified Bank # ☒
New ☐

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☒ 2.0
Comprehension or Analysis ☐

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments: ISDC_CK07.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	3.3	
30	K/A # 004 K2.03 Knowledge of the bus power supplies to the charging pumps.		

Proposed Question:

Operations has aligned Charging Pump P-55C to its alternate power supply to support maintenance on its breaker. If a fault on Bus 1C resulted in a loss of the front bus in the switchyard, which charging pumps would have power?

A.) All three charging pumps P-55A, P-55B, and P-55C
B.) Only charging pumps P-55A and P-55B
C.) Only charging pump P-55C
D.) None of the charging pumps will have power.

Proposed Answer: B

Explanation:
A.) Incorrect - See explanation below.
B.) **Correct** - See explanation below.
C.) Incorrect - See explanation below.
D.) Incorrect - See explanation below.
P-55C is normally powered by LCC-11 off Bus 1C. When on its alternate supply it is powered by LCC-13. LCC-13 is powered by Bus 1C. A fault on Bus 1C that de-energized the front bus would cause Bus 1D and Bus 1E to fast transfer to startup power. P-55C would loss power.

Technical Reference(s): SOP-2A, P&ID's E-3 sheet 1 & E4- Sheet 1

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge ☐
- Comprehension or Analysis ☒ 3.0

10 CFR Part 55 Content: 55.41 b.7
55.43

Comments: ISFB_CK07.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.7	
31	K/A #	004 G 2.3.11 Ability to control radiation releases as it applies to the chemical volume control system.	

Proposed Question:

What is the MINIMUM amount of seal leakage at which SOP-2A, Chemical and Volume Control System, would recommend a Charging Pump be secured?

A.) 401 ml
B.) 801 ml
C.) 1301 ml
D.) 1601 ml

Proposed Answer: C

Explanation:
A.) Incorrect - SOP-2A requires the charging pump to be secured if leakage exceeds 900 ml/min.
B.) Incorrect - SOP-2A requires the charging pump to be secured if leakage exceeds 900 ml/min.
C.) **Correct** - SOP-2A requires the charging pump to be secured if leakage exceeds 900 ml/min.
D.) Incorrect - SOP-2A requires the charging pump to be secured if leakage exceeds 900 ml/min.
Securing a leaking charging pump at the right time will help prevent a radiation release. Failure to take this action can lead to high contamination levels in the auxiliary building and elevated count rates at the plant stack.

Technical Reference(s): SOP-2A

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

Question Source:
Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☒	3.0
	Comprehension or Analysis	☐	

10 CFR Part 55 Content:	55.41	b.10/12
	55.43	

Comments: CVCS_CK16.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.5	
32	K/A # 005 K6.03 Knowledge of the effect of a loss or malfunction on the RHR heat exchanger on the RHRS.		

Proposed Question:

Assume the plant is on Shutdown Cooling when a complete loss of Instrument Air occurs.

Which of the following describes the effect on the Shutdown Cooling System and on the Primary Coolant System (PCS)?

A.) Since CV-3006, SDC Hx Bypass, fails CLOSED, the PCS will begin to heat up.
B.) Since CV-3025, SDC Hx Outlet, fails CLOSED, the PCS will begin to heat up.
C.) Since CV-3006, SDC Hx Bypass, fails OPEN, there is a concern for PCS overcooling.
D.) Since CV-3025, SDC Hx Outlet fails OPEN, there is a concern for PCS overcooling.

Proposed Answer: B

Explanation:
A) Incorrect - The bypass fails open
B) **Correct** - Since CV-3025 is on the outlet of the SDC heat exchanger, and is failed CLOSED, that means there is no component cooling water providing cooling to SDC return to the PCS.
C) Incorrect - With the bypass open there will be less cooling of the PCS.
D) Incorrect - CV-3025 fails closed.

Technical Reference(s): ONP-7.1, and ONP-17

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

Question Source:
- Bank # ☒
- Modified Bank # ☐
- New ☐

Question History: Last NRC Exam 2003

Question Cognitive Level:
- Memory or Fundamental Knowledge ☐
- Comprehension or Analysis ☒ 3.0

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments: SDC_CK21.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	3.1	
33	K/A #	005 A4.04 Ability to manually operate and/or monitor in the control room controls and indications for closed cooling water pumps.	

Proposed Question:

The plant has been tripped and the operators are in the process of working through EOP-4 Loss of Coolant Accident. Safety Injection has actuated and containment pressure is 3 psig and rising, when all off-site power is lost. Following the loss of power both Diesel Generators start and load. When the sequencers have timed out the NCO at the controls notes that CCW P-52C is not running.

Is the response of P-52C correct and why?

A.) No, the pump should have started on Safety injection.
B.) No, the pump should have sequenced on when power was restored.
C.) Yes, P-52C is not safety related and doesn't get a sequencer start signal.
D.) Yes, P-52C only starts if there is a low CCW system pressure.

Proposed Answer: D

Explanation:
A) Incorrect - P-52C doesn't get an auto start in this situation
B) Incorrect - P-52C doesn't start if pressure is normal
C) Incorrect - it gets a start signal if pressure is low.
D) **Correct -** P-52C does get a start signal but only if CCW system pressure is low.

Technical Reference(s): Lesson Plans for CCW and SIS

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☒ 2.5

10 CFR Part 55 Content: 55.41 b.7
55.43

Comments: CCW_CK10.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.8	

34 K/A # 006 K5.02 Knowledge of the operational implications of the relationship between accumulator volume and pressure.

Proposed Question:

Given the following:

- The plant is in MODE 1.
- T-82A, Safety Injection Tank (SIT) sampling is to be performed.
- During the pre-job brief for this evolution, it is discussed that per SOP-3, Safety Injection and Shutdown Cooling System, SIT pressure should not be allowed to lower below 200 psig.

What operational concern is addressed by the above precaution?

A. Will ensure adequate sample flow from the SIT to the sample point.
B. Will ensure SIT pressure remains above the low alarm setpoint.
C. Avoids drawing a vacuum on the SIT during level reduction.
D. Avoids the potential for water hammer upstream of PCS check valves.

Proposed Answer: D

Explanation:

A. Incorrect - There is no minimum pressure requirement for the purpose of ensuring adequate sample flow.
B. Incorrect - The low pressure alarm is expected during SIT sampling. Setpoint is 205 psig
C. Incorrect - For certain plant conditions, there is a concern for a minimum pressure due to a vacuum, but it is misapplied here.
D. **Correct** - Per the technical reference, steam voids can be encountered upstream of the PCS check valves when the plant is at operating temperature, and the check valves leak slightly. The water hammer is a concern if these steam voids collapse during introduction of cooler water which will collapse the voids.

Technical Reference(s): SOP-3, ARP-8

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

Question Source:	Bank #	☐	
	Modified Bank #	☐	
	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☒	3.0
	Comprehension or Analysis	☐	
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments: SIS_CK02.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	4.0	
35	K/A # 006 A3.01 Ability to monitor automatic operation of the ECCS, including the Accumulators.		

Proposed Question:

For a Large Break LOCA (which includes a Loss of All Offsite Power and a double-ended break of the Cold Leg), which of the following represents the expected correct sequence of events?

A. Start of HPSI and LPSI injection
Broken loop SIT starts injecting
Broken loop SIT empties
Remaining SITs start injecting

B. Broken loop SIT starts injecting
Broken loop SIT empties
Start of HPSI and LPSI injection
Remaining SITs start injecting

C. Start of HPSI and LPSI injection
Broken loop SIT starts injecting
Remaining SITs start injecting
Broken loop SIT empties

D. Broken loop SIT starts injecting
Remaining SITs start injecting
Start of HPSI and LPSI injection
Broken loop SIT empties

Proposed Answer: D

Explanation:
A.) Incorrect - Candidate fails to recognize that for the analyzed accident, pumps must be sequenced on (which takes time); in the meantime, analysis shows the SITs will inject first on the broken loop.
B.) Incorrect -
C.) Incorrect -
D.) **Correct** - This is the order of events as described in the FSAR. It is important that operators know the expected response in order to monitor performance.

Technical Reference(s): FSAR Chapter 14, Table 14.17.1-5

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

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☒ 3.0

10 CFR Part 55 Content: 55.41 b.7
55.43

Comments: SIS_CK02.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.6	
36	K/A #	007 A2.02 Ability to (a) predict the impacts of abnormal pressure in the PRT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of abnormal pressure in the PRT.	

Proposed Question:

Following a plant start-up the ATC (at the controls) NCO notes rising level, pressure and temperature trends in the Quench Tank T-73. After entering the appropriate ONP for a primary coolant leak, the crew determines that there is a small leak past one of the Pressurizer relief valves.

If pressure were allowed to rise, at what pressure would the rupture disc fail and what action is directed to reduce/stop the leakage?

A.) 10 psig, vent Quench Tank to containment.
B.) 10 psig, Lower PCS pressure.
C.) 90 psig, vent Quench Tank to the Vent Gas Collection Header.
D.) 90 psig, Lower PCS pressure.

Proposed Answer: D

Explanation:
A) Incorrect, 10 pounds is not the rupture point of the Quench Tank, it is the alarm point.
B) Incorrect, 10 pounds is not the rupture point of the Quench Tank, it is the alarm point.
C) Incorrect, the procedure doesn't direct venting the quench tank to reduce leakage.
D) Correct, 90 psig is the limit, and lowering PCS pressure is what is prescribed by the procedure.

Technical Reference(s): ARP-4, window 32, 46

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

Question Source:
Bank # ☐
Modified Bank # ☐
New ☒

Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.0
	Comprehension or Analysis	☐	
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments: PCS_CK16.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.7	
37	K/A #	008 A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including CCW pressure.	

Proposed Question:

The plant is in Mode 6 with the cavity flooded to 648'. **NO** refueling operations are in progress. CCW Heat Exchanger E-54A has been removed from service to allow for a tube inspection. No other safety related equipment is out of service.

If a loss of off-site power were to occur and all equipment responded as expected what action would the control operators be REQUIRED to take?

A.) Ensure only one Component Cooling Water Pump is operating.
B.) Secure either LPSI Pump P-67A or P-67B.
C.) Secure Auxiliary Feedwater Pumps, P-8A and P-8B.
D.) Open CCW Hx E-54B Service Water Flow Valve CV-0826.

Proposed Answer: A

Explanation: A.) with only one heat exchanger in service the plant is only allowed to operate a single CCW pump to prevent excessive CCW flow through the remaining CCW heat exchanger. The normal shutdown sequencer will start two CCW pumps. As a result one must be secured. B.) neither LPSI pump will start on a normal shutdown sequencer (they only get a start signal on the DBA sequencer) the crew will actually need to start one per ONP-17. C.) Auxiliary Feedwater pumps are given a start permissive. There is no auto start of the auxiliary feedwater pumps on the normal shutdown sequencer. The CCW Hx E-54 V service water flow CV-0826 fails to the open position.

Technical Reference(s): SOP-16 page 14, E-17 sheet 4.

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

Question Source:

- ☐ Bank #
- ☐ Modified Bank #
- ☒ New

Question History: Last NRC Exam ______________

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☒ 3.0

10 CFR Part 55 Content: 55.41 b.5
55.43 ______

Comments: CCW_CK16.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.5	
38	K/A # 010 K2.02 Knowledge of bus power supplies to the controller for PZR spray valve.		

Proposed Question:

The A channel of Pressurizer Pressure control is powered by ....

A.) Y-10
B.) Y-20
C.) Y-30
D.) Y-40

Proposed Answer: A

Explanation: A channel of the pressurizer pressure control is powered by Y-10. This is only a two channel system so the correct answer is not as obvious as this may seem.

Technical Reference(s): ONP-24.1

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

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☒ 2.0
Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 b.7
55.43

Comments: PPCS_CK07.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	3.2	
39	K/A # 010 K6.03 Knowledge of the effect of a loss or malfunction of the PZR sprays and heaters will have on the PZR PCS.		

Proposed Question:

With the plant at full power, a loss of all offsite power occurs. All other plant equipment functions as designed. On the Reactor trip what is the initial effect on Primary Coolant System pressure (pressure at 10 seconds versus pre-trip), and the appropriate method for controlling PCS pressure once entry into EOP-8.0, Loss of Forced Circulation, occurs?

A. pressure is higher; use Auxiliary Spray and Bus 1D heaters to control PCS pressure.
B. pressure is lower, operate Atmospheric Dump Valves to control PCS pressure.
C. pressure is higher; operate Atmospheric Dump Valves to control PCS pressure.
D. pressure is lower, use Auxiliary Spray and Bus 1D heaters to control PCS pressure.

Proposed Answer: D

Explanation:
A) Incorrect, pressure lowers after this reactor trip.
B) Incorrect; candidate incorrectly applies guidance from EOP-3.0 (Station Blackout) that use of ADVs is desired. Despite the loss of offsite power operators still have use of aux spray and 1/2 heaters
C) Incorrect, pressure lowers after this reactor trip. Candidate incorrectly applies guidance from EOP-3.0 (Station Blackout) that use of ADVs is desired. Despite the loss of offsite power operators still have use of aux spray and 1/2 heaters
D) Correct, pressure lowers and the operator can reset Bus 1D heaters and align auxiliary spray.

Technical Reference(s): EOP-8

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge ☒ 2.5
Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 b.7
55.43 ____

Comments: PPCS_CP01.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	3.1	

40 K/A # 012 K5.02 Knowledge of the operational implications of the power density as they apply to the RPS.

Proposed Question:

An ASI alarm will be annunciating for which one of the following conditions?

A.) Core Power 40%, ASI +0.10
B.) Core Power 40%, ASI -0.10
C.) Core Power 100%, ASI +0.10
D.) Core Power 100%, ASI -0.10

Proposed Answer: D

Explanation: This question requires that the candidate have a basic understanding of the shape of the Local Power Density Function that is the main input to the ASI alarm. The correct answer can be reached by recognizing the that top core is more restrictive than bottom core and that the LPD Function is more restrictive at high power.

Technical Reference(s): Core Operating Limits Report (COLR).

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

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☒ 3.0

10 CFR Part 55 Content: 55.41 b.5
55.43

Comments: NI_CK14.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	3.8	
41	K/A #	013 K4.16 Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the avoidance of PTS.	

Proposed Question:

With the plant at 100% power the control room receives both SIRW Tank Hi-Lo temperature alarms. Control room determines that SIRW tank temperature is low at 74°F. An AO sent to the field determines that the SIRW tank heat exchanger's steam side is water logged.

In addition to needing to raise the SIRW tank temperature what other impact does this alarm have on plant operation?

A.) I&C needs to verify that the thermometers are reading accurately since the tanks T.S. limit may be approached.
B.) The tank is potentially reaching the boron precipitation temperature if the tank's boron is near the upper limit.
C.) If SIRW were used to flood the cavity during refueling the vessel minimum temperature would be exceeded.
D.) The tank temperature has reached the point where it may contribute to overcooling the vessel during a SBLOCA or an Excess Steam Demand event.

Proposed Answer: D

Explanation: A.) is wrong because the tech spec is set at 40°F and the alarm setpoint has nothing to do with instrument uncertainties to the tech spec. This was true back when the alarm setpoint was 45°F. B.) There is no potential for boron precipitation at the allowed tank concentrations. This distracter was used because we have been trying to maintain SIRW tank boron concentrations higher to support outage evolutions. C.) There is a concern with respect to a vessel temperature below 73°F but that evolution would be done slowly, and you can't go less than 73F by adding 74F water. D) The minimum temperature of the SIRW tank was changed to reduce the likelihood of a PTS event based on PRA.

Technical Reference(s): DBD 2.01, ARP-8, & SOP-3

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

Question Source: Bank # ☐
Modified Bank # ☐

	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☒	3.5
	Comprehension or Analysis	☐	
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments: SIS_CK16.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.9	

42 K/A # 022 K3.01 Knowledge of the effect that a loss or malfunction of the Containment Cooling System will have on the containment equipment subject to damage by high or low temperature, humidity, and pressure.

Proposed Question:

Consider if the plant were to experience two separate excessive steam demand events inside containment on the A steam generator. During both events the leak size was the same. The only difference between the two events is that service water to containment was secured on the second event.

During the second event, Pressurizer level would indicate ______ and the Steam Generator B level would indicate ______ than during the first event. (assume between 1 to 2 hours into the event)

A.) Higher, Lower
B.) Higher, Higher
C.) Lower, Higher
D.) Lower, Lower

Proposed Answer: B

Explanation: Both of these level instruments have external wet reference legs that are exposed to containment conditions. The higher containment temperature caused by the lack of service water would raise the temperature in these reference legs and lower the waters density. This would cause them to indicate high.

Technical Reference(s): EOP Supplement 9, 10, and 11

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.5

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments: TBAD_TBCORE_CK05.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	4.2	

43 K/A # 026 K1.01 Knowledge of the physical connections and/or cause-effect relationships between the CSS and the ECCS.

Proposed Question:

Given the following conditions:

- A LOCA has occurred inside Containment.
- A Recirculation Actuation Signal (RAS) has been received.
- 2400V Bus 1D equipment is operating.
- 2400 V Bus 1C is de-energized and isolated due to a fault.
- The operators are aligning for recirculation in accordance with EOP-4.0, Loss of Coolant Accident Recovery.

Which of the following alignments would provide the MAXIMUM permissible spray flow and subcooling flow?

A.) ONE Containment Spray Valve open and
ONE HPSI Subcooling Valve open.
B.) ONE Containment Spray Valve open and
BOTH HPSI Subcooling Valves open.
C.) BOTH Containment Spray Valves open and
ONE HPSI Subcooling Valve open.
D.) BOTH Containment Spray Valves open and
BOTH HPSI Subcooling Valves open.

Proposed Answer: A

Explanation:
a. **Correct** - With only 1 CS Pump and 1 HPSI Pump capable of operating, only 1 CS valve and 1 HPSI subcooling valve are permitted to be open.
b. Only 1 HPSI subcooling valve can be open.
c. Only 1 CS valve can be open.
d. Only 1 CS valve and 1 HPSI subcooling valve are permitted to be open.

Technical Reference(s): EOP Supplement 42

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

Question Source:	Bank #	☒	
	Modified Bank #	☐	
	New	☐	
Question History:	Last NRC Exam	2001	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.0
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments: CSS_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	4.2	
44	K/A #	026 K4.01 Knowledge of CSS design feature(s) and/or interlock(s) which provide for the source of water for CSS, including recirculation phase after LOCA.	

Proposed Question:

Which of the following combination of SIRWT levels will provide the required logic to generate a Recirculation Actuation Signal (RAS)?

	LS-0327 (LEFT CHANNEL)	**LS-0328 (RIGHT CHANNEL)**	**LS-0329 (LEFT CHANNEL)**	**LS-0330 (RIGHT CHANNEL)**
A.	1%	5%	5%	1%
B.	1%	5%	1%	5%
C.	5%	1%	5%	1%
D.	5%	1%	5%	5%

Proposed Answer: A

Explanation:
a. **CORRECT** RAS actuation requires either LS-0327 or LS-0329 below 2% AND either LS-0328 or LS-0330 below 2%.
b. Plausible since combination of levels is required for RAS. Either LS-0328 or LS-0330 must also be below 2%.
c. Plausible since combination of levels is required for RAS. Either LS-0327 or LS-0329 must also be below 2%.
d. Plausible since combination of levels is required for RAS. Either LS-0327 or LS-0329 must also be below 2%.

Technical Reference(s): E-17, Sheet 5

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

Question Source: Bank # ☒

	Modified Bank #	☐	
	New	☐	
Question History:	Last NRC Exam	2000	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.0
	Comprehension or Analysis	☐	
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments: CSS_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	3.6	

45 K/A # 039 K3.05 Knowledge of the effect that a loss or malfunction of the MRSS will have on the RCS.

Proposed Question:

Following a refueling outage, the plant is in a Chemistry Hold at 35% power, when the Turbine Bypass Valve CV-0511 fails open.

Assuming no operator action and no change in turbine load, what is the approximate final steady state reactor power level?

A.) 31%
B.) 35%
C.) 39%
D.) 43%

Proposed Answer: C

Explanation:
a.) Candidate erroneously subtracts the 4% from initial power level. b.) Incorrectly believes TBV steam load is not significant enough to cause a change in power. c.) CORRECT. The TBV accounts for steam flow equivalent to ~4% power. d.) Incorrectly doubles TBV 4% load and adds to initial power level.

Technical Reference(s): DBD 1.09, 3.3.1.4

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

Question Source: Bank # ☒ MSS_CK12.0-6
Modified Bank # ☐
New ☐

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☒ 2.0

10 CFR Part 55 Content: 55.41 b.7

55.43 ______

Comments: MSS_CK12.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.9	
46	K/A # 039 G 2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure as it applies to the Main and Reheat Steam System.		

Proposed Question:

Which of the following is **NOT** an action from ONP-23.2 to help limit the spread of contamination following the discovery of a Steam Generator Tube Leak?

A.) Start a plant heating boiler.
B.) Route after condenser drains to the condenser.
C.) Isolate Steam Generator Blowdowns.
D.) Isolate the Main Steam Reheaters.

Proposed Answer: D

Explanation:
A) Incorrect a plant heating boiler is started so that all extraction steam can be isolated.
B) Incorrect after condenser drains can be routed to the floor for chemistry control, the procedure ensures they are captured.
C) Incorrect blowdowns are isolated to keep contamination levels in the turbine building down.
D) Correct, there is no step in the ONP for isolating the reheaters.

Technical Reference(s): ONP-23.2

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

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☒ 3.0
Comprehension or Analysis ☐

10 CFR Part 55 Content:	55.41	b.12
	55.43	

Comments: IOTF2_E13.01

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	3.1	
47	K/A #	059 A4.11 Ability to manually operate and monitor in the control room recovery from automatic feedwater isolation.	

Proposed Question:

With the plant at 70% power the steam flow indication to the 'A' Steam Generator Level controller, FIC-0701, fails HIGH. Operators are alerted to the condition when EK-0961, STEAM GEN E-50A HI LEVEL, alarms. The NCO reports the following conditions.

- A S/G Level 88% and lowering.
- B S/G Level 61% and rising quickly.
- A S/G Main Feed Reg. CV-0701 is closed.
- B S/G Main Feed Reg. CV-0703 is 70% open and closing.
- Both Main Feed Pump speeds are rising slowly.

What actions should the operator take in accordance with the appropriate procedure?

A.) Monitor S/G levels and ensure the level dominate system returns S/G levels to target.
B.) Take manual control of the Main Feed Reg. valves and restore levels to target.
C.) Trip the reactor, the high level override has failed and A S/G level cannot be controlled.
D.) Trip the reactor, B S/G level response is abnormal indicating additional problems.

Proposed Answer: B

Explanation: A) it is possible that the level control system could recover from this failure without operator action, however it would not return level to target. It would restore level to some value above the target. B) is correct per ONP-10. C) is built around a misunderstanding of the automatic setpoint for the high level override. There is a trip requirement at 90% - if the candidate misapplies this requirement they would chose C). A candidate may select D) if they do not recognize how the B steam generator would respond to the event described.

Technical Reference(s): ONP-10, ARP-5

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

Question Source: Bank # ☐
Modified Bank # ☐

	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.6	
48	K/A #	061 K1.09 Knowledge of the physical connections and/or cause-effect relationships between the AFW and the PRMS.	

Proposed Question:

The plant is operating at full power. Auxiliary Feedwater Pump P-8A is out of service for an oil change. Operators start to see a rising trend in RIA-0631, Condenser Off-gas monitor, RIA-0707, Steam Generator Blowdown monitor, and RIA-2323 Main Steam E-50B monitor. After a short time it is determined that the primary to secondary leakage exceeds the trip criteria of ONP-23.2, Steam Generator Tube Leak, and the plant is manually tripped. On the plant trip Bus 1D de-energizes, Diesel Generator 1-2 starts, but doesn't load. The first attempt to close the diesel generator output breaker fails.

Once the crew has transitioned to EOP-5 Steam Generator Tube Rupture, which of the following describes continued operation of Auxiliary Feedwater Pump P-8B?

A.) P-8B is receiving steam from the A Steam Generator and can continue to be operated.
B.) P-8B is receiving steam from the A Steam Generator, but should be secured until faulted generator is isolated.
C.) P-8B is receiving steam from the B Steam Generator and should be secured while P-8A is returned to service.
D.) P-8B is receiving steam from the B Steam Generator but can continue to be run until P-8A is returned to service.

Proposed Answer: A

Explanation:
A.) There is no steam exhaust monitor on the steam driven auxiliary feedwater pump. The original design had the auxiliary feedwater pump being supplied by either generator, but a steam line failure resulted in the supply from the B steam generator being plugged. There is no requirement to secure the steam driven auxiliary feedwater pump in the event of a steam generator tube rupture if the steam driven auxiliary feedwater pump is needed.

Technical Reference(s): EOP-5 step 29 basis, DBD 1.03 page 20
Drawing E-1, sh. 1

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

Question Source: Bank # ☐

	Modified Bank #	☐	
	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.0
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments: TBAF_TBCORE_CK01.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.5	
49	K/A #	061 A3.05 Ability to monitor automatic operation of the AFW, including recognition of leakage, using sump level changes.	

Proposed Question:

Given the following plant conditions:

- The plant is at full power.
- Auxiliary Feedwater Pump P-8A is out of service for maintenance.
- Both Main Feedwater Pumps, P-1A and P-1B spuriously trip.
- The Reactor is then manually tripped and the operators begin performing EOP-1.0, Standard Post Trip Actions.
- AFAS actuates per design.

After transitioning to EOP-2.0, Reactor Trip Recovery, the operator notes that Auxiliary Feedwater flow to BOTH S/Gs quickly lowers to zero.

Assuming no other failures, which one of the following RISING sump level trends is indicative of a single AFW discharge piping leak?

A. Turbine Building sump

B. Containment sump

C. East Safeguards sump

D. West Safeguards sump

Proposed Answer: D

Explanation:
A) Incorrect, P-8A and P-8B are in the turbine building. However for the conditions given P-8C would be the first AFW Pump to start. While the leak would cause P-8B to start the stem says assume no other failures so there would be no leakage in the turbine building.
B) Incorrect, While a single leak in the containment building might result in no flow going to either steam generator there would be a significant flow indicated on the line with the break. The flow indicators are on the line outside of containment. The stem says that both indications go to zero.

C) Incorrect, The candidate believes that P-8C is in east safeguards.
D) Correct. The leak is on the discharge of P-8C so the sump in this room will raise.

Technical Reference(s): ______

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

Question Source:	Bank #	☐	
	Modified Bank #	☐	
	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.0
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.9	

50 K/A # 062 A2.03 Ability to (a) predict the impacts of the consequences of improper sequencing when transferring to or from an inverter on the AC distribution system, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

Proposed Question:

The Preferred AC Bus Y-40 is being re-energized by its inverter. The inverter is loaded prior to opening the CRD Clutch Power Supply (Breaker #11). What impact would this have?

A.) A Reactor trip from CRD power supply current surge.
B.) Overloading of the in-service battery charger (#2 or #4) due to CRD power supply current surge.
C.) Clutch Power Supply insulation breakdown due to CRD power supply current surge.
D.) Overloading of the Preferred AC inverter transformer due to CRD power supply current surge.

Proposed Answer: D

Explanation:
A) incorrect, right idea but this would only cause a half trip not a reactor trip
B) incorrect, this doesn't overload the charger.
C) incorrect, this doesn't damage the insulation, the breaker trips are properly set.
D) correct, per the procedure and plant OE.

Technical Reference(s): SOP-30, 5.4 and section 7.5.1

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

Question Source: Bank # ☒ EPS_CK16.0
Modified Bank # ☐
New ☐

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☒	3.5
	Comprehension or Analysis	☐	

10 CFR Part 55 Content:	55.41	b.5
	55.43	

Comments: EPS_CK16.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.5	

51 K/A # 063 A1.01 Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including battery capacity as it is affected by discharge rate.

Proposed Question:

The plant has tripped from full power due to a loss of all offsite power. NEITHER Diesel Generator will start. For these conditions, which of the following describes the importance of Station Battery Load Stripping, per EOP Supplements 7 and 8?

Station Battery load stripping is designed to ...

A.) extend battery capacity from 30 minutes to two hours while maintaining DC bus voltage at greater than 105 volts.
B.) ensure that the emergency loads are supplied by DC voltage at greater than 120 volts for a minimum of two hours.
C.) extend battery capacity from two hours to four hours while maintaining DC bus voltage at greater than 105 volts.
D.) prevent overheating of the batteries by limiting DC bus amps to less than 50 amps.

Proposed Answer: C

Explanation:
During emergency operation, the station batteries capacity is sufficient to provide 125V DC power to all expected loads (emergency and selected others) for two hours; and still have capacity to close the circuit breakers necessary to restore power to the plant, an original FSAR requirement. The two hour profile can be further extended to four hours by manually stripping the selected loads (See Section 3.2.2).
A) incorrect this action extends performance to 4 hours not 2 hours.
B) the batteries supply 125 volts not 120 volts during the first 2 hours. This has nothing to do with stripping of loads.
D) stripping is never recommended unless amps exceed 157 - 172 amps. 50 amps would be less than expected and indicates candidate doesn't know what the normal accident load is.

Technical Reference(s): DBD 4.01, 3.3.1.4; FSAR 8.4.2.2, EOP Supplement 7/8 basis

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

Question Source: Bank # ☐
Modified Bank # ☐

	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.5
	Comprehension or Analysis	☐	
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments: TBAR_TBCORE_CK02.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	3.2	

52

K/A # 064 K6.08 Knowledge of the effect of a loss or malfunction of the fuel oil storage tanks will have on the ED/G system.

Proposed Question:

Given the following plant conditions:

- The plant has experienced a loss of all offsite power.
- D/G 1-1 is running at a stable load of 2200 KW.
- D/G 1-2 is unavailable.
- T-10A, Fuel Oil Storage Tank, is at its low level alarm.
- There is an unrecoverable leak of 0.7 gpm from T-10A.

For the above conditions, how long will T-10A inventory be able to support D/G 1-1 operation at its current load?

A.) 3.6 days
B.) 5.9 days
C.) 6.5 days
D.) 8.2 days

Proposed Answer: C

Explanation: A.) this answer is calculated assuming two diesels operating. B.) This answer assumes maximum load from the graph not the stated load. C.) is correct 30500 gallons - 1008 gpd leakage, 3714 gpd use. D.) Fails to account for leakage.

Technical Reference(s): SOP-22

Proposed references to be provided to the applicants during examination: SOP-22 Attachments 2 and 3

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☒ 2.5

10 CFR Part 55 Content: 55.41 b.7
55.43

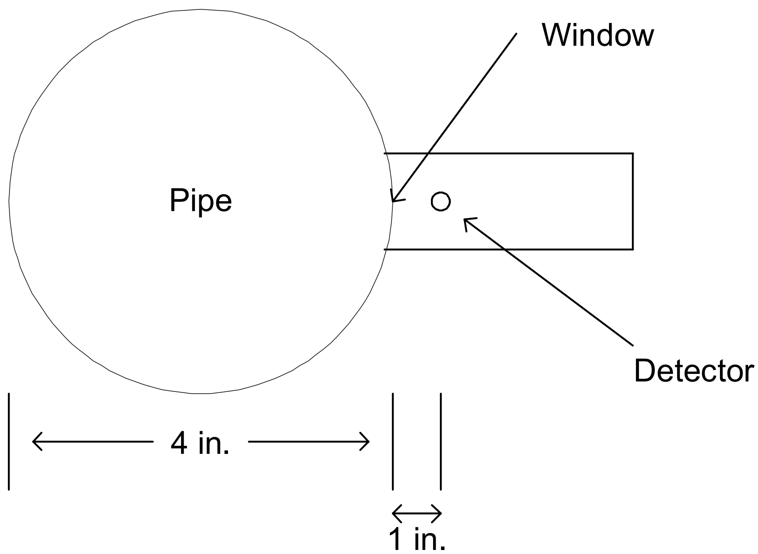
Comments: EDG_CK13.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.5	

53 K/A # 073 K5.02 Knowledge of the operational implications of radiation intensity changes with source distance as they apply to concepts as they apply to PRM system.

Proposed Question:

Assume that a particular waste gas detector consists of a sensing crystal mounted on one side of the pipe with a window through the pipe wall, the pipe is 4 inches in diameter. Assume that the detector is a point detector located 1 inch from the ID of the pipe.



How big of an intensity change can the detector see from a point source (i.e. a hot particle) that passes along the window and one that passes along the far side of the pipe?

A.) A factor of 4.
B.) A factor of 5.
C.) A factor of 16.
D.) A factor of 25.

Proposed Answer: D

Explanation: Dose is proportional to the square of the distance. A count rate of 25 dpm at 1 inch would be equivalent of a count rate of 1 dpm at 5 inches. Wrong answers A) and C) indicate that the candidate did not account for the distance from the edge of pipe to the detector (1 inch). Wrong answer A and B use the wrong proportionality.

Technical Reference(s): GFES Lesson Plan N-RO-01-L-044-I, Radiation Protection

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

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☒ 3.0

10 CFR Part 55 Content: 55.41 b.5
55.43

Comments: GFES

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	2.9	

54 K/A # 076 K4.02 Knowledge of SWS design feature(s) and /or interlock(s) which provide automatic start features associated with SWS pump controls.

Proposed Question:

Non-critical Service Water has been isolated for maintenance and the containment air cooler high capacity valves are closed to help raise containment temperature.

- It is day 40 of an extended plant outage.
- Service Water Pumps P-7A and P-7B are in-service.
- Service Water Pump P-7C is in standby.
- Service water header pressure is 80 psig.

To prevent P-7B from being run at significantly reduced flows, the CRS directs the NCO to secure P-7B. When Service Water Pump P-7B is secured header pressure lowers to 45 psig, Service Water P-7C starts, and header pressure returns to 80 psig.

Should Service Water Pump P-7C have started and why?

A.) Yes, a standby pump starts at 40 psig discharge pressure on either of the other pumps.
B.) Yes, a standby pump starts at 45 psig header pressure.
C.) No, the standby pump should not start when a pump is manually secured.
D.) No, the standby pump should not start until header pressure reaches 40 psig.

Proposed Answer: A

Explanation: The setpoint is 40 psig, however it is not based on header pressure but pump discharge pressure. Each pump has a pressure switch on each of the other pumps discharge. So while header pressure remains above 40 psig, the discharge pressure of P-7B will lower to less than 40 psig and P-7C will start.

Technical Reference(s): ARP-7, SOP-15, P&ID M-213

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

Question Source: Bank # ☐
Modified Bank # ☐

	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.5
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments: SWS_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Importance Rating	3.4	
55	K/A #	078 K3.02 Knowledge of the effect that a loss or malfunction of the IAS will have on systems having pneumatic valves and controls.	

Proposed Question:

An instrument air leak inside containment has prompted the control room to isolate instrument air to containment. Following isolation of instrument air to containment the following alarms are received due to valid signals.

- EK-0706 LETDOWN HX COOLING EXCESS FLOW
- EK-0753 PRESSURIZER PRESSURE OFF NORMAL HI-LO
- EK-0931 PRI COOLANT PUMP P-50A CLG WTR LO FLOW
- EK-1347 CONTAINMENT AIR COOLER SERV WATER LEAK

Which of these alarms is NOT expected based on the air leak and the isolation of instrument air to containment?

A.) EK-0706 LETDOWN HX COOLING EXCESS FLOW
B.) EK-0753 PRESSURIZER PRESSURE OFF NORMAL HI-LO
C.) EK-0931 PRI COOLANT PUMP P-50A CLG WTR LO FLOW
D.) EK-1347 CONTAINMENT AIR COOLER SERV WATER LEAK

Proposed Answer: D

Explanation: A) CV-0909 CCW to the letdown heat exchanger fails open and brings in this alarm. B) Pressurizer Spray valves get their control air from Instrument Air and have no back up. The heaters are energized and the controller has a continuous spray signal in during normal operation to maintain target pressure. C) The excessive CCW flow to the letdown heat exchanger can rob enough flow from the Primary Coolant Pumps to cause this alarm. D) Normal controlled bleedoff flow is isolated by the loss of instrument air and it is redirected to a relief valve via a valve that has a back-up air supply. However in this case instrument air has only been lost inside containment and the normal controlled bleed-off path is not affected.

Technical Reference(s): ARP-4, ARP-5, ARP-8, ONP-7.1

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

Question Source:	Bank #	☐	
	Modified Bank #	☐	
	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.0
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments: IOTF_CK15.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	3.3	

56 K/A # 015 A2.04 Ability to (a) predict the impacts of the effects on axial flux density of control rod alignment and sequencing, xenon production and decay, and boron vs. control rod reactivity changes on the NIS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

Proposed Question:

With the plant at 100% power Control Rod 35 (Group 3 rod very center of the core) drops halfway into the core (66 inches withdrawn). Five minutes after the rod has dropped into the core what will ASI be, how will the NI power indications compare to the actual core power, and what action should be taken to lower core power?

A.) ASI will be positive, NI power will be lower than actual, lower power with group 4 up to 80 inches inserted.
B.) ASI will be positive, NI power will be higher than actual, borate to lower reactor power.
C.) ASI will be negative, NI power will be lower than actual, lower power with group 4 up to 80 inches inserted.
D.) ASI will be negative, NI power will be higher than actual, borate to lower reactor power.

Proposed Answer: B

Explanation: Power will be suppressed in the top of the core. In a CE plant that results in ASI being positive. NI power will be higher than actual because the radial core power profile will have shifted toward the exterior. Xenon effects are minimized by the short time frame of the question (i.e. 5 minutes) Since the NI's are primarily monitoring the exterior of the core they will be at an elevated power compared to actual core power until they are recalibrated. In ONP-5.1 rods are not to be used to lower core power. This reduces the value of the dropped rod and maintains the maximum shutdown margin.

Technical Reference(s): ONP-5.1, P&ID 201 sheet 4

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

Question Source: Bank # ☐

	Modified Bank #	☐	
	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5
10 CFR Part 55 Content:	55.41	b.1	
	55.43		

Comments: IOTF_CK02.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	2.7	
57	K/A # 017 K6.01 Knowledge of the effect of a loss or malfunction of the following ITM system components, sensor and detectors.		

Proposed Question:

If a Core Exit Thermocouple (CET) develops an open circuit in the thermocouple detector, the temperature indication will fail ...

A. high.
B. low.
C. to reference junction temperature.
D. as is.

Proposed Answer: B

Explanation: Because there is no current path the output will fail low.

Technical Reference(s): GFES Lesson Plan, LP N-RO-01-L-020-I, Sensors & Detectors

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

Question Source:
- Bank # ☒ GFES Aug 2005 #P213
- Modified Bank # ☐
- New ☐

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge ☒ 3.0
- Comprehension or Analysis ☐

10 CFR Part 55 Content:
- 55.41 b.7
- 55.43

Comments: GFES

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	3.1	

58 K/A # 027 K5.01 Knowledge of the operational implications of the purpose of charcoal filters as they apply to the CIRS.

Proposed Question:

Which one of the following describes the operation of the containment Iodine Removal Fan units (V-940A/V-940B) and associated charcoal filters?

A.) Automatically started on a Safety Injection Signal (SIAS) to remove I-131 generated in a Loss of Coolant Accident (LOCA).
B.) Manually started during a normal Plant shutdown to remove I-131 for containment habitability.
C.) Manually started during a normal Plant startup to minimize potential I-131 release to the environment.
D.) Automatically start on a Containment High Pressure (CHP) to assist containment Spray System in removing I-131 from containment.

Proposed Answer: B

Explanation:
A) Incorrect, these fans do not have an automatic start signal
B) Correct, fans must be manually started
C) Incorrect, these fans are not run during normal plant startup since there is no one in containment. Not used to mitigate releases in this way.
D) Incorrect, these fans do not have an automatic start signal

Technical Reference(s): GOP-10, SOP-24

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

Question Source: Bank # ☒
Modified Bank # ☐
New ☐

Question History: Last NRC Exam 2003

Question Cognitive Level: Memory or Fundamental Knowledge ☒ 2.0
Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments: IOTA_T07.00

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	2.9	

59 K/A # 029 K3.02 Knowledge of the effect that a loss or malfunction of the Containment Purge System will have on containment entry.

Proposed Question:

During a routine at power containment entry an operator performing QO-5 Containment Isolation Valve Test closes CWRT Vent Valve CV-1064. What impact, if any, does this have on the operator's containment entry?

A.) The operator must don a respirator to continue containment tour.
B.) Dose rates and containment pressure will both start to rise significantly.
C.) Operator will be unable to use the Personnel Airlock due to the differential pressure across the inner door.
D.) Maintenance of a containment vent path is not a condition of the containment entry and its isolation does not effect the entry.

Proposed Answer: D

Explanation: The containment purge system at Palisades is not used except in Mode 5 because the isolation valves are not rated to close against the dP that would be created during an accident. As a result the containment is normally vented through the CWRTs. However this vent path is not large and temporary opening and closing of this path is not significant. As a result our containment entry procedures have been written to compensate for the lack of a containment purge system. A) there is no requirement to have containment vented. B) this is a small path, there would not be a noticeable change in containment conditions. C) the air lock is designed to operate even with large dP's, this would not impact air lock operation.

Technical Reference(s): HP 2.6

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

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.5
	Comprehension or Analysis	☐	

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments: PVT_T03.00

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	2.5	
60	K/A #	034 K1.02 Knowledge of the physical connections and/or cause-effect relationships between the Fuel Handling system and the RHRS	

Proposed Question:

Given the following plant conditions:

- The plant is in MODE 6.
- Fuel movements are in progress.
- Shutdown Cooling is in service with P-67B operating.
- LPSI Pump P-67B unexpectedly trips.
- Reactor cavity level remains stable.

What action is required and why?

A. Immediately attempt one start of P-67A to maintain decay heat removal.

B. Within ONE hour install the Spent Fuel Pool South Tilt Pit gate to preserve cooling to the Spent Fuel Pool.

C. Raise reactor cavity water level by two feet to raise suction pressure to P-67B and attempt one restart of P-67B.

D. Stop ALL fuel movements immediately because of concerns for loss of heat removal and potential radiation hazard.

Proposed Answer: D

Explanation:
A) Incorrect, Per ONP-17 the alternate LPSI pump cannot be started until the cause of the trip is known.
B) Incorrect, There is a requirement to place the tilt pit gate in if refueling operations are going to be stopped for twenty four hours - but the requirement is not that it be installed in one hour nor is it done to isolate the SFP cooling system.
C) Incorrect, There are steps to raise PCS (cavity) level in ONP-17 but they do not apply to this situation since the Pool will be near the upper limit of the cavity level to support refueling. This action would overfill the cavity.
D) Correct, this is the immediate action per the procedure.

Technical Reference(s): ONP-17, Loss of SDC

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

Question Source: Bank # ☐
Modified Bank # ☐
New ☒

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☒ 2.5
Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 b.13
55.43

Comments: IOTF_CK15.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	3.4	
61	K/A #	035 G 2.4.11 Knowledge of the abnormal condition procedures as they apply to the Steam Generators.	

Proposed Question:

Given the following plant conditions:

- Plant is at 80% power and was performing a power escalation to full power when EK-1364, "GASEOUS MONITORING HI RADIATION" annunciated.
- It is determined that this alarm is due to RIA-0631, Condenser Off-Gas Monitor in an alarm condition.
- PCS total gas activity is 0.17 μCi/cc.
- Off Gas flow is 3 cfm.
- At 0610 RIA-0631 indicated 7.00 E3 cpm.
- At 0635 RIA-0631 indicates 1.00 E4 cpm.
- "B" Steam Generator is the affected generator.

What actions should be taken to address the above plant conditions?

A.) Trip the reactor and carry out the Immediate Actions of EOP-1.0, "Standard Post-Trip Actions"
B.) Plant management must evaluate the need to perform a controlled Plant shutdown per GOP-8, "Power Reduction and Plant Shutdown".
C.) Place the Plant in Mode 3 within 6 hours per ONP-23.2, "Steam Generator Tube Leak", Step 4.2.
D.) Place the Plant in Mode 3 within 2 hours per ONP-23.2, "Steam Generator Tube Leak", Step 4.2.

Proposed Answer: D

Explanation:

a. Candidate misinterprets/miscalculates and arrives at an incorrect leak rate, or misuses the decision table based on rate of rise of leak rate.
b. Candidate misinterprets/miscalculates and arrives at an incorrect leak rate, or misuses the decision table based on rate of rise of leak rate.
c. Candidate misinterprets/miscalculates and arrives at an incorrect leak rate, or misuses the decision table based on rate of rise of leak rate.
d. CORRECT - Tube leak rate at 0610 = 0.0412 gpm. Tube leak rate at 0635 = 0.0541 gpm. This indicates a rate of rise of 0.031 gpm / hr which requires a plant shutdown within 4 hours.

Technical Reference(s): ONP-23.2, Steam Generator Tube Leak

Proposed references to be provided to the applicants during examination: ONP-23.2 Step 4.1.h and ATT. 1 and 2

Question Source: Bank # ☒
Modified Bank # ☐
New ☐

Question History: Last NRC Exam 2003

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☒ 3.0

10 CFR Part 55 Content: 55.41 b.10
55.43

Comments: IOTF_CK05.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	2.7	
62	K/A # 041 A4.02 Ability to manually operate and/or monitor in the control room the cooldown valves. (ADVs/TBV)		

Proposed Question:

Given the following plant conditions:

- The plant was at full power when the Reactor tripped.
- Atmospheric Dump Valve (ADV) controller, HIC-0780A, is in AUTO.
- Turbine Bypass Valve (TBV) controller, PIC-0511, is in AUTO.
- Main Condenser vacuum has reduced to 4".
- Instrument Air system pressure has reduced to 20 psig and is stable.

Which one of the following describes the response of the ADVs and the TBV, and what action will the operators take for PCS heat removal?

A.) BOTH the TBV and the ADVs will close; operators will take action to remove PCS heat using the Hogging Air Ejector.
B.) BOTH the TBV and the ADVs will close; operators will take action to remove PCS heat using the once-through-cooling-method.
C.) ONLY the TBV will close; operators will take action to remove PCS heat using the ADVs.
D.) ONLY the ADVs will close; operators will take action to remove PCS heat using the TBV.

Proposed Answer: C

Explanation:

a. ADVs have nitrogen backup supply.
b. ADVs have nitrogen backup supply.
c. Correct - TBV fails closed on loss of air, and does not operate if condenser vacuum is less than 5".
d. ADVs have nitrogen backup supply.

Technical Reference(s): ONP-7.1, Loss of Instrument Air

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

Question Source:	Bank #	☒	
	Modified Bank #	☐	
	New	☐	
Question History:	Last NRC Exam	2001	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments: MSS_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	3.8	
63	K/A #	045 A1.05 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system control including expected response of primary plant parameters (temperature and pressure) following T/G trip.	

Proposed Question:

The grid has experienced tornado damage but the Palisades switchyard is intact. The plant was at 100% power and a loss of all offsite power occurs. The plant responds as follows:

- The Main Turbine automatically trips.
- The Reactor automatically trips.
- The Main Generator automatically trips.
- Immediately after trip, status of the Main Generator relays is as follows:
    - 386Primary (386P) NOT Actuated
    - 386Backup (386B) NOT Actuated
    - 386Coastdown (386C) ACTUATED

Which one of the following describes the significance of the above relay status?

A.) Since neither the 386P nor the 386B relay actuated, the Main Generator has experienced damage due to at least 10 seconds of motorizing.
B.) Because the 386C actuated, autostarting of Bearing Lift Pumps occurred, and protected the bearings from damage.
C.) The 386P should have actuated. Because it did not, PCS temperature and pressure will be not meet acceptance criteria of EOP-1.0, Standard Post Trip Actions.
D.) Because the 386C actuated, decay heat removal and development of natural circulation have been enhanced.

Proposed Answer: D

Explanation:
A. It is expected for the conditions given that neither 386P or 386B would actuate, as these are relays associated directly with faults on the Main Generator. There was no such fault for these conditions. Since the Main Generator did automatically trip, motorizing did not occur.
B. Actuation of 386C should have occurred, however, its actuation actually aids in the

development of natural circulation. C. The 386P should NOT have actuated for the given conditions. D. CORRECT - 386C coastdown relay maintains inertially produced electrical power to the PCPs for 10 seconds, for the above conditions.

Technical Reference(s): DBD 3.03, section 3.2.1.5, 3.2.2.4, Appendix C, page 1 of 2

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge ☐
- Comprehension or Analysis ☒ 3.5

10 CFR Part 55 Content:
- 55.41 b.5
- 55.43

Comments: SPS_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	3.4	
64	K/A #	068 K4.01 Knowledge of design feature(s) and/or interlock(s) which provide for the safety and environmental precautions for handling hot, acidic, and radioactive liquids.	

Proposed Question:

With the plant in MODE 1, the following valid alarm is received:

- EK-1368, RADWASTE PANEL C-40 OFF NORMAL

The AO reports that a non-critical heat trace point associated with the discharge of Recycled Boric Acid Pump P-96 is reading LOW at 150°F. For this heat trace point , the RED light is lit, and the AMBER light is off.

Is the alarmed heat trace circuit energized, and what is the effect on the system?

A.) Yes. Temperature is expected to recover.
B.) Yes. Cannot transfer REBAT until the alarm is clear.
C.) No. Only the redundant channel of heat trace is available on this circuit.
D.) No. need to station portable heaters to prevent boron precipitation in these lines.

Proposed Answer: C

Explanation:
A) Incorrect, the amber light should be lit indicating the circuit is energized.
B) Incorrect, the amber light should be lit indicating the circuit is energized.
C) The primary circuit is not working correctly (yellow light not lit) have to monitor secondary and get primary fixed.
D) Incorrect, this action would only be needed if the secondary heat trace circuit also failed.

Technical Reference(s): ARP-8, EK-1368; ARP-9, SOP-2A, Att. 1

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

Question Source: Bank # ☐
Modified Bank # ☐

	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.5
	Comprehension or Analysis	☐	
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments: CVCS_CK09.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Importance Rating	2.6	

65

K/A # 075 K2.03 Knowledge of bus power supplies to the emergency/essential SWS pumps.

Proposed Question:

During a loss of off-site power event only Diesel Generator 1-1 starts and loads onto its safety bus. Which service water pump(s) is/are available?

A.) P-7A
B.) P-7A and P-7C
C.) P-7B
D.) P-7A and P-7B

Proposed Answer: C

Explanation: Service water P-7B is powered by Bus 1C (D/G 1-1), Service water P-7A and P-7C are power by Bus 1D (D/G 1-2). Since the stem states that Bus 1D is deenergized only P-7B has power.

Technical Reference(s): P&ID, E-3, sh. 1

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

Question Source:
- Bank # ☐
- Modified Bank # ☒
- New ☐

Question History: Last NRC Exam 2003

Question Cognitive Level:
- Memory or Fundamental Knowledge ☒ 2.0
- Comprehension or Analysis ☐

10 CFR Part 55 Content:
- 55.41 b.5
- 55.43

Comments: ISDA_CK07.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	2.8	
66	K/A #	G 2.1.22 Ability to determine Mode of Operation.	

Proposed Question:

Given the following plant conditions:

- PCS average temperature: 425°F
- All control rods: Fully inserted
- PCS cooldown rate: 30°F/hour

What is the current plant mode as defined in Palisades Technical Specifications for these conditions?

A.) MODE 1
B.) MODE 2
C.) MODE 3
D.) MODE 4

Proposed Answer: C

Explanation:
See reference provided.

Technical Reference(s): Tech. Specs. definition section

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

Question Source: Bank # ☒ APTS_E01.04-1
Modified Bank # ☐
New ☐

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ☒ 2.0
Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 b.10

55.43 ______

Comments: APTS_E01.02

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	3.9	
67	K/A #	G 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operations.	

Proposed Question:

Excerpt from GOP-2 Mode 5 to Mode 3 ≥ 525°F

		Time	Date	Initial
5.10	WHEN PCS pressure is at 1700 psia, THEN **VERIFY** SIAS unblocked (Annunciators EK-1337, EK-1338, EK-1339, and EK-1369 clear).	_______	______	_____
5.11	WHEN associated Steam Generator pressure is greater than 550 psia on 2 of 4 safety channels, THEN **PERFORM** the following to verify MSIV closure unblocked:			
	a. **PUSH** HS/LPE-50A AND **VERIFY** EK-0970 clear.	_______	______	______
	b. **PUSH** HS/LPE-50B AND **VERIFY** EK-0970 clear.	_______	______	______

To perform the operation of verifying that MSIV closure is UNBLOCKED, how are the above procedures to be implemented?

A.) You must EXIT GOP-2 and go to SOP-7 to perform the unblocking.
B.) You REMAIN in GOP-2 and refer to SOP-7 to perform the unblocking.
C.) Unblocking is performed per GOP-2 only. Use of SOP-7 is NOT required.
D.) Unblocking is performed per SOP-7 only. Use of GOP-2 is NOT required.

Proposed Answer: C

Explanation:

a. There is NO reference or direction to either REFER TO or GO TO the SOP.
b. There is NO reference or direction to either REFER TO or GO TO the SOP.
c. CORRECT - The details of the step are given in the GOP; no use of the SOP is required. Also, SOP-7 makes no reference to this operation.

d. SOP-7 makes no reference to this operation. More importantly, the GOP is the governing document and clearly prescribes required actions for the operation.

Technical Reference(s): GOP-2, Att.1, step 5.11

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

Question Source:
- Bank # ☒
- Modified Bank # ☐
- New ☐

Question History: Last NRC Exam 2003

Question Cognitive Level:
- Memory or Fundamental Knowledge ☒ 2.0
- Comprehension or Analysis ☐

10 CFR Part 55 Content:
- 55.41 b.10
- 55.43

Comments: IOTA_T02.00

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	3.4	
68	K/A # G 2.1.29 Knowledge of how to conduct and verify valve lineups.		

Proposed Question:

Given the following conditions:

- The main flow through a pipe in a safety-related system at normal pressure is 100 gpm.
- A vent valve on the pipe will allow 4 gpm if full open at normal pressure.
- A drain valve on the pipe will allow 8 gpm if full open at normal pressure.

Which of the following describes the procedural requirements for locking devices for these valves?

	VENT VALVE	**DRAIN VALVE**
A.)	Lock Required	Lock Required
B.)	Lock Required	Lock NOT Required
C.)	Lock NOT Required	Lock Required
D.)	Lock NOT Required	Lock NOT Required

Proposed Answer: C

Explanation:

a. Plausible if candidate determines that all vent and drain valves require locks regardless of flow value. Requirement is based on percentage of flow.
b. Plausible if candidates determines all vent valves require locks and drain valves do not. Requirement is based on percentage of flow.
c. CORRECT Below 5% of flow a lock is not required. Above 5% flow a lock is required. Vent valve does not require lock, but drain valve does. Plausible if candidate determines value was greater than 5%. Limit is 5% of main flow.
d. Plausible if candidate determines value was greater than 5%. Requirement is based on 5% of main flow.

Technical Reference(s): AP 4.02

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

Question Source:	Bank #	☒	2001 Cert.
	Modified Bank #	☐	
	New	☐	

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments: No specific training objective found

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	3.6	
69	K/A #	G 2.2.13 Knowledge of tagging and clearance procedures.	

Proposed Question:

Refer to the following list of valve operations:

1. Close discharge valve.
2. Close suction valve.
3. Open discharge valve.
4. Open suction valve.

Which of the following describes the required sequence of valve operations when tagging out and subsequently restoring to service of a centrifugal pump?

	TAGOUT		RESTORE
A.)	1,2	then	4,3
B.)	2,1	then	4,3
C.)	1,2	then	3,4
D.)	2,1	then	3,4

Proposed Answer: A

Explanation:
A) Correct, close discharge before suction, and open suction before discharge.
B) Incorrect, isolates suction first.
C) Incorrect, opens discharge first.
D) Incorrect, order for both tagout and restore wrong.

Technical Reference(s): FP-OP-TAG-01, Admin Proc. 4.18

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

Question Source:
- Bank # ☒
- Modified Bank # ☐
- New ☐

Question History: Last NRC Exam 2003

Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.0
	Comprehension or Analysis	☐	

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments: TAGGING

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	2.6	
70	K/A # G 2.2.23 Ability to track limiting conditions of operations.		

Proposed Question:

A Technical Specification Action has the following requirements for completion time:

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3 AND B.2 Be in MODE 5	6 hours 36 hours

This Action is entered at 0735 on Tuesday, and the plant enters MODE 3 at 1115 the same day.

Which ONE of the following is the latest time by which the plant must be in MODE 5?

A.) 1715 Wednesday
B.) 1935 Wednesday
C.) 2315 Wednesday
D.) 0135 Thursday

Proposed Answer: B

Explanation:
A) Incorrect, candidate has allowed only 30 hours from time of entry into mode 3.
B) Correct, 0735 plus 36 hours is 1935 on Wednesday.
C) Incorrect, candidate has allowed for 36 hours from the time of entry into mode 3.
D) Incorrect, candidate has allowed 42 hours from the time of entry into condition.

Technical Reference(s): Tech Spec

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

Question Source: Bank # ☒ APTS_E01.08-1

	Modified Bank #	☐	
	New	☐	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments: APTS_E.01.08

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	2.5	
71	K/A #	G 2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	

Proposed Question:

An auxiliary operator will be working out on the reactor head with an NCO during control rod coupling and testing. The auxiliary operator has received a total dose of 1890 mrem YTD. An ALARA evaluation has been done and it has been decided that the operators will NOT have respirator protection. The average dose rate on the head is 27 mrem/hr and 5 DAC.

How long can the AO support the NCO before reaching his annual administrative exposure limit?

A.) 1 hour 5 min
B.) 2 hours 45 min
C.) 3 hours 25 min
D.) 4 hours 5 min

Proposed Answer: B

Explanation: 1 DAC-hr is equal to 2.5 mrem. The administrative annual dose limit is 2000 mrem. The wrong answers are based on using 5/1 1/1 and 0/1 ratios. The 0/1 ratio would be selected if the candidate did not think that DAC counted against the dose limits.

Technical Reference(s): HP 8.9, Admin Proc. 7.04, Att. 1

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

Question Source:	Bank #	☐	
	Modified Bank #	☐	
	New	☒	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	3.0

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments: GAT

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	2.9	
72	K/A # G 2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.		

Proposed Question:

Following a forced shutdown to repair a leaking CRDM seal, it is desired to perform ESSO-8, Shutdown Cooling Heat Exchanger Radiation Level Reduction, which involves flushing the SDC Heat exchanger with the SIRW Tank. Which of the following conditions will allow performance of this procedure?

ADVs = Atmospheric Dump Valves
MO-3015/3016 = Shutdown Cooling Water Return Valves.

A.) MODE 4, ADVs available, MO-3015/3016 CLOSED.
B.) MODE 5, ADVs isolated, MO-3015/3016 OPEN.
C.) MODE 4, ADVs isolated, MO-3015/3016 OPEN.
D.) MODE 5, ADVs available, MO-3015/3016 CLOSED.

Proposed Answer: A

Explanation:
A) Correct, Plant needs to be in Mode 4 (to allow PCS to be cooled by S/G), ADVs must be available, and shutdown cooling must be isolated.
B) Incorrect, SDC would be needed to cool PCS in mode 5. Candidate may have misconception of ESSO-8.
C) Incorrect, SDC needs to be isolated in order for the heat exchangers to be filled with low activity water.
D) Incorrect, SDC would be needed to cool PCS in mode 5.

Technical Reference(s): ESSO-8, 3.2.a and 3.3.d

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

Question Source:
- Bank # ☐
- Modified Bank # ☐
- New ☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.5
	Comprehension or Analysis	☐	

10 CFR Part 55 Content:	55.41	b.12
	55.43	

Comments: SDC_CK01.0

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	3.1	
73	K/A # G 2.4.7 Knowledge of event based EOP mitigation strategies.		

Proposed Question:

Given the following conditions:

- A Steam Generator Tube Rupture has occurred on SG 'A'.
- SG 'A' has been isolated.
- HPSI Pumps have been secured.
- A cooldown is in progress using SG 'B' ADVs.
- 'B' S/G pressure is currently at 650 psia.
- SG 'A' indicated level is at 142% and slowly rising.
- SG 'A' pressure is 840 psia.
- Pressurizer pressure is 920 psia.
- Pressurizer level is 44%.
- All PCPs are secured.

For these conditions Pressurizer pressure should be ...

A.) REDUCED to less than 790 psia to ensure main steam code safeties remain closed.
B.) REDUCED to less than 840 psia to establish backflow from SG 'A' to the PCS.
C.) RAISED to at least 970 psia to maintain adequate subcooling.
D.) RAISED to at least 940 psia to prevent backflow dilution of the PCS.

Proposed Answer: B

Explanation:
A.) Temperature is reduced to less than 524 to prevent main steam code safeties from lifting, this corresponds to 840 psia not 790 psia.
B.) CORRECT
C.) The affected S/G level is to be maintained less than 140%. Backflow must be established. Raising Pressurizer pressure would raise the rate of S/G level rise. The 'B' S/G that is be used for PCS cooling indicates adequate subcooling.
D.) Raising Pressurizer pressure to at least 940 psia would indeed prevent backflow to the PCS; however, per the EOP basis document, the amount of dilution would not jeopardize shutdown margin.

Technical Reference(s): EOP-5.0, SGTR, step 35

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

Question Source:	Bank #	☒	
	Modified Bank #	☐	
	New	☐	
Question History:	Last NRC Exam	1999	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments: TBAF_E03.03

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	3.3	
74	K/A # 2.4.13 Knowledge of crew roles and responsibilities during EOP flowchart use.		

Proposed Question:

Administrative Procedure 4.06, Attachment 15, EOP Performance Standards, directs that during the performance of an Emergency Operating Procedure the crew should conduct a crew brief prior to which of the following evolutions and why?

A.) Securing Aux. Feedwater to a faulted Steam Generator to discuss the effect on heat removal.
B.) Manually initiating Safety Injection to discuss the effect on cooldown rate and pressure control.
C.) Placing Control Room HVAC to the Emergency Mode to discuss effect on diesel loading.
D.) Throttling Safety Injection to discuss the effect on heat removal and on pressure control.

Proposed Answer: D

Explanation: Answers A, B, and C are all significant events that are not listed in the admin as requiring a crew brief and for which a brief is not expected.

Technical Reference(s): AP 4.06, Att. 15

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

Question Source:	Bank #	☐
	Modified Bank #	☐
	New	☒

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☒	2.5
	Comprehension or Analysis	☐	

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments: TBAA_E05.01

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	Importance Rating	3.5	
75	K/A # 2.4.46 Ability to verify alarms are consistent with the plant conditions.		

Proposed Question:

The following plant conditions exist:

- Reactor Power 100%.
- Instrument Air Compressors C-2A in HAND.
- Instrument Air Compressor C-2B in AUTO.
- Instrument Air Compressor C-2C is OOS for maintenance.
- The Instrument Air system experiences a transient that momentarily dropped pressure in the header to 86 psig. Pressure recovers within two (2) minutes and stabilizes at 99 psig.
- Control Room alarm EK-1105, "AIR COMPRESSORS STANDBY COMP RUNNING" is in alarm.
- Instrument Air Compressor, C-2B "RED" indicating light above the control switch is on.

Ten minutes after C-2B started, the Auxiliary Operator calls to report that Instrument Air Compressor, C-2B is not running.

This condition is present due to the fact that the Instrument Air Compressor, C-2B ...

A.) started in the loaded condition and therefore tripped on overcurrent.
B.) started and cycled on and off to maintain header pressure between 85-100 psig.
C.) started unloaded and therefore tripped on low lube oil pressure.
D.) ran unloaded for more than 6 minutes and then automatically shut down.

Proposed Answer: D

Explanation:
A) Incorrect, loading is controlled by the start circuitry and doesn't cause a trip.
B) Incorrect, these compressors do not shutdown when maintaining pressure in the required band.
C) Incorrect, the low lube oil pressure trip is also controlled by the start circuitry.
D) Correct, since it was started on a transient and then the load returned to normal, C-2A was able to maintain load. When this happens C-2B goes into 'sleep' mode as described in the answer.

Technical Reference(s): SOP-19, step 7.2.2.c, Note prior to; SOP-19, Att. 6, p.2, step j.

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

Question Source:	Bank #	☒	CAS_CK14.0-1
	Modified Bank #	☐	
	New	☐	

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	☐	
	Comprehension or Analysis	☒	2.5

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:
CAS_CK10.0