(Test Key) 2023 NRC Byron SRO Exam

Byron ILT NRC/Cert Examination (Master)

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Test	2023 NRC Byron SRO Exam
VISION ID	371408
Status	

EXAMINATION COVER SHEET

Exam Title (ID)	2023 NRC Byron SRO Exam (371408)			
Training Program	Byron ILT NRC/Cert Examination (Master)			
LMS Component ID		**Total Points**	100.00	**Pass Criteria** ≥ 80 %
Trainee Name		**Employee ID**		
Graded By / Date		**Grade**	____ / 100.00 = _____ %	

Review and Approval			
Instructor		**Date**	
Technical Review		**Date**	
Training Supv		**Date**	

Examination Rules

1. References may NOT be used during this exam, unless otherwise stated.
2. Read each question carefully before answering. If you have any questions or need clarification during the exam, contact the exam proctor.
3. Conversation with other trainees during the exam is prohibited.
4. Partial credit will NOT be considered, unless otherwise stated. Show **all** work and state **all** assumptions when partial credit may be given.
5. Restroom trips are limited and only one examinee at a time may leave.
6 . For exams with time limits, you have ____ minutes to complete the exam.
7. The examinee agrees to refrain from discussing the content of the exam until the end of the exam cycle.

Examination Integrity Statement

Cheating or compromising the exam will result in disciplinary actions up to and including termination.

“I acknowledge that I am aware of the Exam Rules stated above. Further, I have not given, received, or observed any aid or information regarding this exam prior to or during its administration that could compromise this exam.”

Examinee Signature _____ Date _____

Review Acknowledgement

“I acknowledge that the correct answers to the exam questions were indicated to me following the completion of the exam. I have had the opportunity to review the exam questions with the instructor to ensure my understanding.”

Examinee Signature _____ Date _____

Question 1	ID: 2504046	Points: 1.00

- Unit 2 had an inadvertent reactor trip from 100% power.
- All systems operate as designed.
- The RO is maintaining S/G levels per 2BEP-0, REACTOR TRIP OR SI, step 17, CHECK SG LEVELS.

Per 2BEP-0, the MINIMUM AF flow should be maintained > (1) GPM until at least one SG NR level is > (2) .

A. (1) 500
(2) 10%

B. (1) 500
(2) 14%

C. (1) 900
(2) 10%

D. (1) 900
(2) 14%

Answer	B

Answer Explanation

A – Plausible: Because 10% minimum level is correct for Unit 1.

B – Correct: Per 2BEP-0 step 17 and CAS, AF flow should be maintained >500 GPM until at least one SG NR level is >14%.

C – Plausible: Because 900 GPM is correct for maintaining SG levels in 1BFR-S.1 and 10% minimum level is correct for Unit 1.

D – Plausible: Because 900 GPM is correct for maintaining SG levels in 1BFR-S.1.

Question Information

Topic	Byron 2023 NRC Exam Q01				
User ID	RE10007-N06			System ID	2504046
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	Difficulty	
Technical Reference and Revision #	2BEP-0 Rev. 305		
Training Objective	T.EP01-05		
Previous Exam Use	None		

References Provided	None
K/A Justification	Question meets the K/A, requires examinee knowledge of the operational implications of the AF system decay heat removal during a reactor trip recovery.
SRO-Only Justification	N/A
Additional Information	The question is Tier 1 because it requires knowledge to of an EOP CAS.

K/A Reference(s)

EPE.007.EK1.06	Safety Function	Tier 1	Group	RO Imp: 3.9	SRO Imp:
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to a Reactor Trip: (CFR: 41.8 / 41.10 / 45.3) Decay heat removal capability of AFW					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 2 ID: 2496470 Points: 1.00

- 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, is in progress.
- A RCS pressure transient caused PZR pressure to rise to 2400 psig.
- 1RY455A, PZR PORV is 100% open.
- 1RY456, PZR PORV is in the intermediate position and stuck open.
- PRT pressure is 50 psig.

(1) Currently the steam entering the PRT is _____.

(2) When PZR pressure drops, what is the MAXIMUM PZR pressure that the RO will close 1RY8000B, PORV ISOL VLV, to isolate 1RY456?

A. (1) saturated
(2) 2315 psig

B. (1) saturated
(2) 2345 psig

C. (1) superheated
(2) 2315 psig

D. (1) superheated
(2) 2345 psig

Answer	A

Answer Explanation

A – Correct: Part 1: Convert question stem pressures from psig to psia, resulting in RCS pressure of 2415 psia and PRT pressure of 65 psia. Using the Mollier Diagram, plot where 2415 psia intersects the saturation line (at approximately 1110 BTU/lb. enthalpy). Then move directly to the right (along the enthalpy line) because the throttling process is isenthalpic (constant enthalpy) until intersecting the 65 psia constant pressure curve (PRT pressure). Note that the intersection point is below the saturation curve in the saturated steam section.
Part 2: 1BEP-1, step 5 (CAS) checks pressurizer PORVs closed. If they are not, it instructs the RO to close or isolate any PORV when pressure is less than 2315 psig.

B – Plausible: Because 2345 psig is the initial opening setpoint of 1RY455A. However, in automatic the PORV does not fully close until 2315 psig.

C – Plausible: Because any combination of PZR pressure and PRT pressure that would result in the isenthalpic throttling process ending above the saturation curve would result in superheated steam entering the PRT.

D – Plausible: Because 2345 psig is the initial opening setpoint of 1RY455A. However, in automatic the PORV does not fully close until 2315 psig. Also because any combination of PZR pressure and PRT pressure that would result in the isenthalpic throttling process ending above the saturation curve would result in superheated steam entering the PRT.

Question Information

Topic	Byron 2023 NRC Exam Q02				
User ID	RE10008-N06			**System ID**	2496470
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	Steam Table, Mollier Diagram 1BEP-1 Rev. 302		
Training Objective	ELO 2.4 (193003)		
Previous Exam Use	None		

References Provided	None
K/A Justification	Question meets the K/A, requires examinee knowledge of the operational implications of a open PZR PORV.
SRO-Only Justification	N/A
Additional Information	The question is Tier 1 and RO license level because it requires knowledge of emergency procedure OAS.

K/A Reference(s)

APE.008.AK1.01	Safety Function	Tier 1	Group	RO Imp: 4.1	SRO Imp:
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to a Pressurizer Vapor Space Accident: (CFR: 41.8 / 41.10 / 45.3) Thermodynamics and flow characteristics of open or leaking PZR PORV or safety valves					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 3 ID: 2496504 Points: 1.00

- Unit 1 was at 100% power when an RCS LOCA occurred.
- RCS pressure is <1425 psig.

In accordance with 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1 CHECK IF RCPS SHOULD BE STOPPED, why are RCPs tripped ? (assuming other required conditions are met)

A. Continued operation of the RCPs would increase the mass being lost through the break.

B. RVLIS will provide erratic indication if the RCPs continue to operate.

C. The RCPs are not effective in cooling the core if pumping a two-phase mixture.

D. The RCP seals will be subjected to severe damage if continued RCP operation is allowed.

Answer	A

Answer Explanation

A - Correct: The WOG GENERIC ISSUE RCP TRIP/RESTART basis document states that if RCPs trip at an unfavorable time during a SB LOCA, more mass is lost and core is uncovered more deeply.

B - Plausible: Because RVLIS indication is not used as a emergency procedural criteria with RCPs running but the RCPs are not tripped in order to use RVLIS.

C - Plausible: Because core cooling is less effective if two-phase but RCPs will pump a two-phase mixture.

D - Plausible: Because RCPs are tripped on a Phase B isolation which is expected during a Large Break LOCA because of a loss of CC cooling to oil coolers and thermal barriers, but seal cooling is maintained during a SB LOCA.

Question Information

Topic	Byron 2023 NRC Exam Q03				
User ID	BYLC3DEP02B006			**System ID**	2496504
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from Byron LOR bank	**Difficulty**	
Technical Reference and Revision #	BEP-1 WOG Background Rev.300 BD WOG GENERIC ISSUE RCP TRIP/RESTART Rev. 3		
Training Objective	T.EP01-02		
Previous Exam Use	None		

References Provided	None
K/A Justification	Question meets K/A – the candidate must know the relationship between the RCPs and a SB LOCA.
SRO-Only Justification	N/A
Additional Information	The question is Tier 1 because it requires knowledge of emergency procedure step bases.

K/A Reference(s)

EPE.009.EK2.05	Safety Function	Tier 1	Group	RO Imp: 3.7	SRO Imp:
Knowledge of the relationship between a Small-Break LOCA and the following systems or components: (CFR: 41.7 / 45.7) RCPS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 4	ID: 2496508	Points: 1.00

- Unit 1 is at 26% power.
- The 1A RCP breaker tripped open due to undervoltage on Bus 157.

What automatic or required manual action will occur as a result of the trip of the RCP?

A. A normal plant shutdown will be initiated.

B. The reactor will be manually tripped by the operator.

C. The reactor will automatically trip due to the open RCP breaker.

D. The reactor will automatically trip due to RCS loop low flow condition.

Answer	B

Answer Explanation

A - Plausible: Because the single loop loss of flow trip is blocked below P-8 (30% power). This would be correct if reactor power was >30%.

B - Correct: No AUTO trip is expected due to power < P-8. Administrative direction for a RCP trip in these conditions is to trip reactor. BAR 1-13-A3, operator actions directs "IF in MODE 1 or 2, and RCP breaker has opened or low flow conditions (<90%) exist, THEN VERIFY/INITIATE reactor trip and go to 1BEP-0, Reactor Trip or Safety Injection Unit 1."

C - Plausible: Because the RCP Breaker Open trip requires 2 of 4 breakers open to generate a trip below P-8.

D - Plausible: Because if a single RCP tripped in Mode 3, the crew response may be to perform a normal plant cooldown. However, the procedural direction in this situation is to trip the reactor.

Question Information

Topic	Byron 2023 NRC Exam Q04				
User ID	NRC 2014 4			System ID	2496508
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from 2014 Byron NRC	Difficulty	

Technical Reference and Revision #	BAR 1-BP-3.7 Rev. 1 BAR 1-13-A3 Rev. 4
Training Objective	T.EP01-06-A
Previous Exam Use	None

References Provided	None
K/A Justification	Candidate must have ability to monitor and operate the plant during a RCP malfunction when the high power reactor trip interlock is blocked.
SRO-Only Justification	N/A
Additional Information	The question is Tier 1 because it requires procedural directed operator action to be taken.

K/A Reference(s)

APE.015.AA1.15	Safety Function	Tier 1	Group	RO Imp: 3.4	SRO Imp:
Ability to operate and/or monitor the following as they apply to Reactor Coolant Pump Malfunctions: (CFR: 41.7 / 45.5 / 45.6) High-power/low-flow reactor trip block status					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 5	ID: 2496651	Points: 1.00

- Unit 1 is at 100% power.
- A loss of charging through the regen heat exchanger occurred.
- 1BOA PRI-15, CVCS ABNORMALITIES, is in progress.
- Charging has been minimized to flow only through the RCP seals.
- Letdown has been isolated.
- VCT level is 30% and slowly dropping.
- VCT makeup can NOT be established.

In accordance with 1BOA PRI-15, the crew will establish Excess Letdown to ...

A. maintain RCS chemistry with the mixed bed demineralizers.

B. prevent VCT level dropping to 0% and CV pump cavitation.

C. prevent an uncontrolled PZR level drop.

D. prevent an uncontrolled PZR level rise.

Answer	D

Answer Explanation

A - Plausible: Because normal letdown will maintain RCS chemistry by maintaining flow through the mixed bed letdown demineralizers. However, this flow path is not available from Excess Letdown operation.

B - Plausible: Because VCT level is dropping without RMCS available. However, if VCT level continues to drop to 5%, the VCT level control system will automatically open 1CV112D/E and stop the VCT level drop before the CV pumps cavitate.

C - Plausible: Because there is an imbalance between charging and letdown affecting PZR level.

D - Correct: 1BOA PRI-15 establishes Excess Letdown immediately after isolating normal Letdown. With the question stem conditions, PZR level will be rising due to charging through the RCP seals with no letdown flow. Therefore, Excess Letdown is necessary to prevent PZR level from rising uncontrollably.

Question Information

Topic	Byron 2023 NRC Exam Q05				
User ID	RE10022-N06			**System ID**	2496651
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A

Technical Reference and Revision #	1BOA PRI-15, Rev. 4
Training Objective	T.OA23A-04
Previous Exams Use	None

References Provided	None
K/A Justification	The question meets the K/A by requiring knowledge of reason for establishing excess letdown with a loss of reactor coolant makeup.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring knowledge of reason for step in the site's AOP procedure. RO license level because the question requires understanding of the procedure's purpose and overall mitigative strategy.

K/A Reference(s)

APE.022.AK3.03	Safety Function	Tier 1	Group	RO Imp: 3.3	SRO Imp:
Knowledge of the reasons for the following responses and/or actions as they apply to Loss of Reactor Coolant Makeup: (CFR: 41.5 / 41.10 / 45.6 / 45.13) Establishing excess letdown					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 6 ID: 2496653 Points: 1.00

Which of the following plant conditions are ALL used in 1BOA PRI-10, LOSS OF RH COOLING, Table A, DETERMINATION OF EFFECTIVE RCS COOLING METHODS?

1. Containment is ADVERSE (whether or not)
2. Plant Tech Spec MODE (current status)
3. RCS is ADVERSE (whether or not)
4. RCS in INTACT (whether or not)

A. 1, 2 and 3

B. 1, 2 and 4

C. 2, 3 and 4

D. 1, 3 and 4

Answer	C

Answer Explanation

A - Plausible: Containment adverse conditions are used in multiple emergency procedures to determine procedural criteria for maintaining plant parameters and decision points. However, containment adverse conditions are not used in 1BOA PRI-10, Table A.

B - Plausible: Containment adverse conditions are used in multiple emergency procedures to determine procedural criteria for maintaining plant parameters and decision points. However, containment adverse conditions are not used in 1BOA PRI-10, Table A.

C - Correct: 1BOA PRI-10, Table A, uses plant Mode, and RCS status (intact and adverse conditions) to determine which alternate RCS cooling method is effective during a loss of RH.

D - Plausible: Containment adverse conditions are used in multiple emergency procedures to determine procedural criteria for maintaining plant parameters and decision points. However, containment adverse conditions are not used in 1BOA PRI-10, Table A.

Question Information

Topic	Byron 2023 NRC Exam Q06				
User ID	RE10025-N04			**System ID**	2496653
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Modified from Byron ILT bank question BYLITOA43005	**Difficulty**	N/A
Technical Reference and Revision #	1BOA PRI-10, Rev. 108		
Training Objective	T.OA20-03		
Previous Exams Use	None		

References Provided	None
K/A Justification	The question meets the K/A by requiring knowledge of reason for which alternate RCS heat removal method is selected during a loss of RH event.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring knowledge of reason for step in the site's AOP procedure. RO license level because the question requires understanding of the procedure's purpose and overall mitigative strategy.

Original question:

Per BOA PRI-10, LOSS OF RH COOLING, which ONE of the following is **NOT** a consideration for determining if the RCS is adverse?

A. All RCS loops BLOCKED

B. Containment pressure HIGH

C. Existence of a cold leg OPENING

D. Reactor vessel head INSTALLED

Answer	B

K/A Reference(s)

APE.025.AK3.01	Safety Function	Tier 1	Group	RO Imp: 3.8	SRO Imp:
Knowledge of the reasons for the following responses and/or actions as they apply to the Loss of the Residual Heat Removal System: (CFR: 41.5 / 41.10 / 45.6 / 45.13) Shift to alternate flowpath					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 7 ID: 2496655 Points: 1.00

- An ATWS has occurred on Unit 2.
- U2 RCS pressure is currently 2360 psig and stable.
- 2BFR-S.1, "Nuclear Power Generation/ATWS", is in progress.

With the above conditions, U2 RCS pressure must be LOWERED to...

A. reduce the pressure/void coefficient to add negative reactivity.

B. ensure sufficient charging flow for boric acid addition.

C. reduce hydraulic pressure to facilitate control rod insertion.

D. ensure adequate safety injection flow to make up for RCS inventory losses.

Answer	B

Answer Explanation

A - Plausible: Because as pressure increases, density increases, increasing the moderator-to-fuel ratio. In the under-moderated core, the increase in the moderator-to-fuel ratio results in positive reactivity addition. Therefore, the pressure coefficient is a positive reactivity coefficient. Additionally, the void coefficient would add negative reactivity to the core as pressure is reduced.

B - Correct: Per BD-FR-S.1, The check on RCS pressure is intended to alert the operator to a condition which would reduce CENT CHG pump injection into the RCS and, therefore, boration. The PZR PORV pressure setpoint is chosen as that pressure at which flow into the RCS is insufficient.

C - Plausible: Because control rods insert during a reactor trip into a pressurized reactor vessel. Several factors influence the insertion times of the control rods including mechanical friction and hydraulic friction. Although the CRDMs are also at RCS pressure, to a novice applicant, reducing RCS pressure may facilitate the rod insertion process.

D - Plausible: Because reducing RCS pressure is a mitigation strategy for loss of core cooling (depressurizing SGs to depressurize the RCS and restore ECCS injection) or for loss of secondary heat sink (opening PZR PORVs to facilitate ECCS injection during RCS bleed and feed).

Question Information

Topic	Byron 2023 NRC Exam Q07				
User ID	RE10029-N06			**System ID**	2496655
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Modified from Bwd ILT bank question RE10029-003	**Difficulty**	N/A
Technical Reference and Revision #	BD-FR-S.1, Rev. 300		
Training Objective	T.FR1-04		
Previous Exams Use	None		

References Provided	None
K/A Justification	The question meets the K/A by requiring knowledge of reason for emergency boration substep during an ATWS.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring knowledge of reason for step in the site's AOP procedure. RO license level because the question requires understanding of the procedure's purpose and overall mitigative strategy.

Original question:

An ATWS has occurred on Unit 2.

- RCS pressure is currently 2360 psig.
- 2BwFR-S.1, "Nuclear Power Generation/ATWS", is in progress.

RCS pressure must be LOWERED to:

A. ensure sufficient charging flow for boric acid addition.

B. below the Safety reliefs setpoints to prevent a challenge to RCS integrity.

C. below the PZR PORV setpoints to avoid a rupture of the PRT.

D. ensure adequate ECCS injection to make up for RCS inventory losses.

Answer	A

K/A Reference(s)

EPE.029.EK3.11	Safety Function	Tier 1	Group	RO Imp: 4.3	SRO Imp:
Knowledge of the reasons for the following responses and/or actions as they apply to an Anticipated Transient Without Scram: (CFR: 41.5 / 41.10 / 45.6 / 45.13) Initiating emergency boration					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 8 ID: 2496729 Points: 1.00

- A SGTR has occurred on Unit 1.
- The crew is performing 1BEP-3, STEAM GENERATOR TUBE RUPTURE.
- The RCS has been cooled down to the target temperature per 1BEP-3.
- The crew is currently depressurizing the RCS with maximum pressurizer spray valves.

The RCS is being depressurized to...

A. raise ECCS flow.

B. raise RCS subcooling.

C. minimize PTS concerns.

D. minimize ruptured SG break flow.

Answer	D

Answer Explanation

A - Plausible: Because reducing RCS pressure is a mitigation strategy for loss of core cooling (depressurizing SGs to depressurize the RCS and restore ECCS injection) or for loss of secondary heat sink (opening PZR PORVs to facilitate ECCS injection during RCS bleed and feed).

B - Plausible: Because establishing RCS subcooling margin is also a high level action of 1BEP-3. However, raising subcooling is a function of the RCS cooldown (not the RCS depressurization).

C - Plausible: Because PTS concerns will be reduced due to the depressurization. However, 1BEP-3 contains a caution prior to initiating the RCS cooldown that PTS indications may be false in the affected loop (if RCPs are not running) and should be disregarded until after the cooldown and ECCS is terminated.

D - Correct: Per BEP-3 high level action summary, the RCS is depressurized to minimize break flow and restore RCS inventory.

Question Information

Topic	Byron 2023 NRC Exam Q08				
User ID	RE10038-N05			**System ID**	2496729
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BEP-3 Rev. 306		
Training Objective	T.EP04-01		
Previous Exams Use	None		

References Provided	None
K/A Justification	The question meets the K/A by requiring knowledge of relationship between SGTR and the PZR PCS.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring knowledge of high level action summary of EP procedure.

K/A Reference(s)

EPE.038.EK2.15	Safety Function	Tier 1	Group	RO Imp: 3.6	SRO Imp:
Knowledge of the relationship between a Steam Generator Tube Rupture and the following systems or components: (CFR: 41.7 / 41.8 / 45.4 / 45.7 / 45.8) PZR PCS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 9 ID: 2496740 Points: 1.00

- Unit 1 reactor was tripped from 100% power.
- 1BEP-0, REACTOR TRIP OR SI, is in progress.

The following Unit 1 SG parameters are present 30 seconds after the reactor trip:

STEAM GENERATOR	1A	1B	1C	1D
• Steam Pressure (psig)	900	900	900	1025
• MSIV	OPEN	OPEN	OPEN	CLOSED
• NR Level	5%	0%	20%	40%
• MSL Radiation Monitor	GREEN	GREEN	YELLOW	GREEN
• Steam Flow (mlbm/hr)	0.1	0.1	0.1	0.0

The following Unit 1 SG parameters are present 30 SECONDS LATER:

STEAM GENERATOR	1A	1B	1C	1D
• Steam Pressure (psig)	900	900	900	1025
• MSIVs	CLOSED	CLOSED	CLOSED	CLOSED
• NR Level	5%	0%	25%	40%
• MSL Radiation Monitor	GREEN	GREEN	YELLOW	GREEN
• Steam Flow (mlbm/hr)	0.4	0.0	0.0	0.0

Based on the above conditions, which Unit 1 SG is FAULTED?
(NOTE: Assume ONLY 1 SG is faulted.)

A. 1A

B. 1B

C. 1C

D. 1D

Answer	A

Answer Explanation

A - Correct: 1A SG is showing steam flow with its MSIV closed and below the SG PORV setpoint.

B - Plausible: Because 1B SG has the lowest NR level which may be the result of a faulted SG. However, low SG level can be a result of other events (such as imbalanced FW flow).

C - Plausible: Because 1C SG has highest MSL rad monitor indication which may be indicative of a ruptured main steam line in the area of the rad monitor. However, elevated steam line rads are more likely the result of a SG tube leak.

D - Plausible: Because the 1D SG had the MSIV closed prior to the other 3 SGs and has the highest pressure. However, the 1D SG pressure did not exceed the PORV lift setpoint of 1115 psig.

Question Information

Topic	Byron 2023 NRC Exam Q09				
User ID	RE10040-N05			**System ID**	2496740
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

NRC Exams Only			
Question Type	Bank from Bwd LOR bank	**Difficulty**	N/A
Technical Reference and Revision #	ILT Lesson Plan N-BY-TQ-MS-23 Rev. 11		
Training Objective	T.EP03-02		
Previous Exams Use	None		

References Provided	None
K/A Justification	The question meets the K/A by requiring ability to determine steam line rupture from pressure and flow indicators.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems.

K/A Reference(s)

APE.040.AA2.01	Safety Function	Tier 1	Group	RO Imp: 3.6	SRO Imp: 4.0
Ability to determine and/or interpret the following as they apply to a Steamline Rupture: (CFR: 41.10 / 43.5 / 45 .13) Occurrence and location of a steamline rupture from pressure and flow indications					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 10	ID: 2501592	Points: 1.00

- Unit 1 experienced a loss of Main Feedwater and Auxiliary Feedwater and has subsequently entered 1BFR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.
- The crew is performing step 11, TRY TO ESTABLISH CONDENSATE BOOSTER FLOW TO AT LEAST ONE SG and is assessing the SGs status for DRY conditions.
- SG levels are dropping.
- Containment pressure is 6.0 psig.

With the above conditions, the HIGHEST indicated level that would meet the criteria as a DRY SG per 1BFR-H.1, is level LESS THAN...

A. 31% NARROW range.

B. 10% NARROW range.

C. 27% WIDE range.

D. 10% WIDE range.

Answer	C

Answer Explanation

A - Because the 1BST-3 criteria for loss of heat sink and entry conditions into 1BFR-H.1 is NARROW RANGE < 10% (31% ADVERSE CNMT) with no feed flow. This answer is less than the setpoint for loss of heat sink with no feed flow with adverse containment.

B - Plausible: Because this answer is less than the setpoint for loss of heat sink, with no feed flow, with normal containment.

C - Correct: Per 1BFR-H.1, A dry SG is any SG with WIDE RANGE level less than 10% (27% ADVERSE CNMT), and no feed flow established. Because containment is adverse, <27% WR is the answer that meets the criteria.

D - Plausible: Because this answer would be correct if containment was not adverse.

Question Information

Topic	Byron 2023 NRC Exam Q10				
User ID	RE10054-N01			**System ID**	2501592
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Modified from Byron 2022 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BFR-H.1, Rev.304		
Training Objective	T.FR03-04-A		
Previous 2 Byron NRC Exams Use	Byron 2022 NRC exam		

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of EOP note.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's EOP procedure.

Original Question:

- Unit 1 experienced a loss of Main Feedwater and Auxiliary Feedwater and has subsequently enter 1BFR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.
- The crew is performing step 11, TRY TO ESTABLISH CONDENSATE BOOSTER FLOW TO AT LEAST ONE SG and is assessing the SGs status for DRY conditions.
- SG levels are dropping.
- Containment pressure is 4.0 psig.

With the above conditions, the HIGHEST indicated level that would meet the criteria as a DRY SG per 1BFR-H.1, is level LESS THAN...

A. 31% NARROW range.

B. 10% NARROW range.

C. 27% WIDE range.

D. 10% WIDE range.

Answer	**D**

K/A Reference(s)

P2.4.20	Safety Function	Tier	Group	RO Imp: 3.8	SRO Imp: 4.3
Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions, and notes (CFR: 41.10 / 43.5 / 45.13)					
G.APE.054	Safety Function	Tier 1	Group	RO Imp:	SRO Imp:
Loss of Main Feedwater					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 11 ID: 2496811 Points: 1.00

- Unit 1 is in Mode 1.
- Unit 2 is in Mode 6 with the reactor cavity FILLED.
- The Unit 1 FC Cooling Pump is OOS.

A loss of offsite power occurs on Unit 2.
- Power is restored to all Unit 2 4KV buses.
- The Unit 2 FC Cooling Pump FAULTS during the electrical transient.
- The crew enters 0BOA REFUEL-3, LOSS OF SPENT FUEL COOLING.
- Per 0BOA REFUEL-3, alternate cooling to the Spent Fuel Pit is aligned.

With the current Spent Fuel Pit cooling alignment, the 2FH001, UNIT 2 XFER TUBE ISOL VLV, is ___(1)___ and FC cooling flow is being provided by ___(2)___ .

A. (1) OPEN to connect the Fuel Pool to the U-2 Reactor Cavity
(2) a FC Purification pump

B. (1) OPEN to connect the Fuel Pool to the U-2 Reactor Cavity
(2) the FC Skimmer pump

C. (1) CLOSED to isolate the Fuel Pool from the U-2 Reactor Cavity
(2) a FC Purification pump

D. (1) CLOSED to isolate the Fuel Pool from the U-2 Reactor Cavity
(2) the FC Skimmer pump

Answer	A

Answer Explanation

A - Correct: Per 0BOA REFUEL-3 (note prior to step 7), the alternate FC cooling flow path is:

-Rx cavity
-Purification pump
-FC HX
-Spent Fuel Pool
-Sluice Gate and Transfer Tube
-Rx Cavity

This is an abnormal alignment that is only used for alternate cooling per 0BOA REFUEL-3. Therefore, the 1FH001 must be open (allowing flow through the transfer tube and into the refuel cavity) and flow is being provided by the FC Purification pump.

B - Plausible: Because the FC Skimmer pump is the only other available pump that has a direct suction from the fuel pool. However, the skimmer loop only has a flow path through a filter but has no cooling capabilities.

C - Plausible: Because normal alignment of a Purification Pump does not go through the 1FH001 valve. The purification loop normal alignment is, pump suction from the RWST or Rx cavity, flow through the FC demins and return back to the RWST or Rx cavity.

D - Plausible: Because normal alignment of a Purification Pump does not go through the 1FH001 valve nor through the FC Hx. The purification loop normal alignment is, pump suction from the RWST or Rx cavity, flow through the FC demins and return back to the RWST or Rx cavity. Also because the FC Skimmer pump is the only other available pump that has a direct suction from the fuel pool. However, the skimmer loop only has a flow path through a filter but has no cooling capabilities.

Question Information

Topic	Byron 2023 NRC Exam Q11				
User ID	RE10056-N07			**System ID**	2496811
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	0BOA REFUEL-3 Rev. 4		
Training Objective	T.OA31-03		
Previous 2 Byron NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of the operational implications of long term fuel pool cooling following a LOOP.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's AOP procedure and the assessment of the integrated plant response to an abnormal situation crossing several plan systems

K/A Reference(s)

APE.056.AK1.08	Safety Function	Tier 1	Group	RO Imp: 3.6	SRO Imp:
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of Offsite Power: (CFR: 41.8 / 41.10 / 45.3) Long-term spent fuel pool cooling					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 12 ID: 2496817 Points: 1.00

- Both Units are at 100% power, normally aligned.
- A loss of offsite power occurs on Unit 1 ONLY.
- Simultaneously, ALL DC control power is LOST to the 1A DG (NO OTHER EQUIPMENT IS AFFECTED).
- Subsequently, the crew enters 1BOA ELEC-3, LOSS OF 4KV ESF BUS.

(1) The 1A DG will _____.

(2) Per 1BOA ELEC-3, power will be restored to Bus 141 from the _____.

A. (1) NOT start
(2) 2A DG

B. (1) NOT start
(2) Unit 2 SATs

C. (1) start, but NOT energize bus 141
(2) 2A DG

D. (1) start, but NOT energize bus 141
(2) Unit 2 SATs

Answer	B

Answer Explanation

A - Plausible: Because, if the LOOP had occurred on both units or if the loss of all AC had occurred, bus 141 could have been fed from the 2A DG. However, 1BOA ELEC-3 crossties bus 141 to whichever power source is currently supplying bus 241.

B - Correct: A loss of control power to a DG will trip the DG if previously running and prevent the DG from starting. Per 1BOA ELEC-3, any ESF bus not energized will be crosstied to the opposite units ESF bus being fed by either the SATs or DG. Because Unit 2 did not lose offsite power and was normally aligned prior to the event, bus 141 would be fed from the Unit 2 SATs.

C - Plausible: Because some diesel engines do not require independent control power to start (example the diesel AF pump engines). Additionally, with a loss of DC control power, the DG would lose field flash for generator excitation current, preventing a running DG from obtaining proper voltage to synch onto a bus. Also, if the LOOP had occurred on both units or if the loss of all AC

had occurred, bus 141 could have been fed from the 2A DG. However, 1BOA ELEC-3 crossties bus 141 to whichever power source is currently supplying bus 241.

D - Plausible: Because some diesel engines do not require independent control power to start (example the diesel AF pump engines). Additionally, with a loss of DC control power, the DG would lose field flash for generator excitation current, preventing a running DG from obtaining proper voltage to synch onto a bus.

Question Information

Topic	Byron 2023 NRC Exam Q12				
User ID	RE10058-N03			**System ID**	2496817
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA ELEC-3 Rev. 109		
Training Objective	T.OA03-03		
Previous 2 Byron NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of relationship between a loss of DC power and the DG.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's AOP procedure. RO license level because the question requires understanding of the procedure's purpose and overall mitigative strategy.

K/A Reference(s)

APE.058.AK2.06	Safety Function	Tier 1	Group	RO Imp: 4.1	SRO Imp:
Knowledge of the relationship between Loss of DC Power and the following systems or components: (CFR: 41.7 / 45.7) EDGs					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 13	ID: 2499746	Points: 1.00

- Unit 1 is at 100% power.
- The crew is adjusting 1SX007, CC HX SX OUTLET VLV, to lower CC outlet temperature from the CC heat exchanger.
- A control circuit malfunction causes 1SX007 to fully CLOSE and its breaker trips.
- The EO is directed to manually reopen 1SX007 with the manual operator wheel.

What is the MAXIMUM CC HX outlet temperature that the crew should maintain CC temperature below during NORMAL plant operations?

A. 128°F

B. 120°F

C. 105°F

D. 96°F

Answer	C

Answer Explanation

A – Plausible: Because the note prior to step 4 in 1BOA PRI-6 allows CC Hx outlet temperature to rise to 128°F during a post LOCA condition.

B – Plausible: Because the note prior to step 4 in 1BOA PRI-6 allows CC Hx outlet temperature to rise to 120°F during a the first 3 hours of an RCS cooldown.

C – Correct: Per 1BOA PRI-6, 1BOA PRI-7 and 1BOP SX-T4 CC Hx outlet temperature maximum is 105°F during normal operations.

D – Plausible: Because 96°F is the Tech Spec limit for SX pump discharge temperature per TS 3.7.9.

Question Information

Topic	Byron 2023 NRC Exam Q13				
User ID	RE10062-N05			**System ID**	2499746
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	**Difficulty**	
Technical Reference and Revision #	BOP SX-T4, Rev. 21 1BOA PRI-6, Rev. 114 1BOA PRI-7, Rev. 111		
Training Objective	T.OA17-03		
Previous Exam Use	None		

References Provided	None
K/A Justification	The question meets the K/A because the candidate must be able to determine the normal upper limit for CC heat exchanger outlet temperature during a loss of SX event.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring knowledge from normal and abnormal operating procedures.

K/A Reference(s)

APE.062.AA2.04	Safety Function	Tier 1	Group	RO Imp: 3.0	SRO Imp: 3.1
Ability to determine and/or interpret the following as they apply to Loss of Service Water: (CFR: 43.5 / 45.13) The normal values and upper limits for the temperatures of the components cooled by SWS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 14	ID: 2496834	Points: 1.00

- Both units are at 100% power in normal alignment.
- The LEAD Station Air compressor TRIPS.
- The LAG Station Air compressor is running with SA and IA pressures slowly DROPPING.
- The crew enters 0BOA SEC-4, LOSS OF INSTRUMENT AIR.
- Current plant air pressures are:
  - SA header pressure - 107 psig
  - IA header pressure - 101 psig

With CURRENT air system pressures...

A. BOTH STDBY SACs have failed to auto start, the crew should MANUALLY start a STDBY SAC.

B. STDBY-1 SAC (ONLY) has failed to auto start, the crew should MANUALLY start a STDBY SAC.

C. STDBY-2 SAC (ONLY) has failed to auto start, the crew should MANUALLY start a STDBY SAC.

D. NEITHER STDBY SAC should have auto started, the crew should MANUALLY start a STDBY SAC IF pressures continue to DROP.

Answer	D

Answer Explanation

A – Plausible: Because instrument air header pressure has dropped below the equivalent station air header pressure auto start setpoint of both standby SACs.

B – Plausible: Because the standby SACs are programmed for a staggered low pressure start per BOP SA-12.

C – Plausible: Because the standby SACs are programmed for a staggered low pressure start per BOP SA-12.

D – Correct: The LAG compressor should maintain SA pressure at approximately 116 psig. The STDBY-1 SAC is programmed to auto start at station air header pressure of 104 psig. The STDBY-2 SAC is programmed to auto start at station air header pressure of 102 psig. Therefore, neither SAC has reached the station air auto start setpoint. BAR 0-37-C2 and 0BOA SEC-4, directs the crew to start a standby SAC if station air pressure drops to 106 psig (BAR 0-37-C2) or 104 psig (0BOA SEC-4).

Question Information

Topic	Byron 2023 NRC Exam Q14				
User ID	RE10065-N03			**System ID**	2496834
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only

Question Type	New for Byron 2023 NRC exam	**Difficulty**	
Technical Reference and Revision #	0BOA SEC-4 Rev. 110 BOP SA-12 Rev.39 BAR 0-37-C2 Rev. 8		
Training Objective	T.OA39-03		
Previous Exam Use	None		

References Provided	None
K/A Justification	The question meets the K/A because it requires candidate ability to operate or monitor service air compressors during a loss of instrument air event.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's AOP and BAR procedures.

K/A Reference(s)

APE.065.AA1.07	Safety Function	Tier 1	Group	RO Imp: 3.3	SRO Imp:
Ability to operate and/or monitor the following as they apply to Loss of Instrument Air: (CFR: 41.7 / 45.5 / 45.6) Instrument and/or service air compressors					

Learning Objective(s)

2023 NRC RO Exam

User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 15	ID: 2496837	Points: 1.00

- Unit 1 is at 100% power.
- A grid disturbance has lowered ESF bus 141 & 142 voltages.
- The ESF buses de-energize, and the DGs both start and re-energize the ESF buses.
- All 4kv ESF equipment sequences on the ESF buses as designed.
- The Control rods step out and annunciator 1-10-D5, BANK D ROD STOP C-11, ALARMS.

With the above conditions, CBD is currently at _(1)_ steps withdrawn.

To clear alarm 1-10-D5, the crew must insert rods to _(2)_ steps withdrawn.

A. (1) 223
(2) 220

B. (1) 223
(2) 222

C. (1) 228
(2) 220

D. (1) 228
(2) 222

Answer	A

Answer Explanation

A – Correct: Per BAR 1-10-D6, C-11 setpoint is 223 steps on CBD and the reset is < 221 on CBD group 1.

B – Plausible: Because 222 steps on CBD is the second highest DRPI indication.

C – Plausible: Because 228 steps is the upper most DRPI indication.

D – Plausible: Because 228 steps is the upper most DRPI indication and 222 steps on CBD is the second highest DRPI indication.

Question Information

Topic	Byron 2023 NRC Exam Q15				
User ID	RE10077-N05			System ID	2496837
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	Difficulty	

Technical Reference and Revision #	BAR 1-10-D5 Rev. 3
Training Objective	S.RD1-14
Previous Exam Use	None

References Provided	None
K/A Justification	The question meets the K/A because it requires candidate ability to operate or monitor the reactor controls following a grid disturbance event.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring analysis of the progression of an event; and the assessment of the integrated plant response abnormal situations and how to correct the event.

K/A Reference(s)

APE.077.AA1.04	Safety Function	Tier 1	Group	RO Imp: 3.6	SRO Imp:
Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 / 41.10 / 45.5 / 45.7 / 45.8) Reactor controls					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 16 ID: 2496850 Points: 1.00

- An RCS LOCA has been confirmed.
- The crew has entered 1BEP-0, REACTOR TRIP OR SAFETY INJECTION.

Which of the following is a subsequent direct entry condition into 1BCA-1.2, LOCA OUTSIDE CONTAINMENT?

A. RCS pressure LESS than 325 psig.

B. UNPLANNED lowering of containment pressure

C. Auxiliary Building sump level alarms that FAIL to clear

D. Auxiliary Building radiation trends ABNORMAL for plant conditions

Answer	D

Answer Explanation

A – Plausible: Because RCS pressure less than 325 psig is an entry condition into 1BEP-1 from 1BEP-0. But not a direct entry into 1BCA-1.2.

B – Plausible: Because an unplanned containment pressure drop is a criteria for a loss of containment in EP-AA-1002, BYRON EAL ANNEX.

C – Plausible: Because an Aux Bldg sump level alarm that fails to clear is a criteria for entry into 1BOA PRI-8, AUXILIARY BUILDING FLOODING.

D – Correct: Per 1BEP-0 and 1BEP-1, Aux Bldg rads not normal for plant conditions is an entry condition into 1BCA-1.2.

Question Information

Topic	Byron 2023 NRC Exam Q16				
User ID	RE1WE04-N03			**System ID**	2496850
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	
Technical Reference and Revision #	1BCA-1.2 Rev. 300		
Training Objective	T.CA2-02		
Previous Exam Use	None		

References Provided	None
K/A Justification	The question meets the K/A because it requires knowledge of LOCA outside containment procedure entry conditions.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring knowledge of EOP procedure.

K/A Reference(s)

P2.4.02	Safety Function	Tier 3	Group	RO Imp: 4.5	SRO Imp: 4.6
Knowledge of system setpoints, interlocks, and automatic actions associated with emergency and abnormal operating procedure entry conditions (CFR: 41.7 / 45.7 / 45.8)					

G.W.E04	Safety Function	Tier	Group	RO Imp:	SRO Imp:
LOCA Outside Containment					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 17 ID: 2496853 Points: 1.00

- An RCS LOCA occurred on Unit 1.
- 1SI8811A/B, CNMT SUMP ISOL VLVs, are CLOSED and CANNOT be opened.
- 1BCA-1.1 "LOSS OF EMERGENCY COOLANT RECIRCULATION" was entered.
- Minimum ECCS flow to remove decay heat has been established per 1BCA-1.1.

With the above conditions, 1BCA-1.1 requires that the RO maintain a MINIMUM primary plant inventory of...

A. 4% in the Pressurizer.

B. 12% in the Pressurizer.

C. 31% in RVLIS HEAD region.

D. 15% in RVLIS PLENUM region.

Answer	D

Answer Explanation

A – Plausible: Because 4% pressurizer level is criteria for determining SI actuation criteria in 1BEP ES-0.2.

B – Plausible: Because 12% pressurizer level is criteria for determining if ECCS flow should be reduced in 1BEP-0

C – Plausible: Because 31% in the RVLIS head region would represent the first indication that reactor vessel inventory was not completely full.

D – Correct: 1BCA-1.1 Continuous Action Summary requires the operator to raise RCS make-up flow if either the CETCs rise or if the RVLIS plenum region drops below 15%. This is less restrictive than criteria used in other emergency procedures to raise ECCS flow because of the objective to conserve RWST water.

Question Information

Topic	Byron 2023 NRC Exam Q17				
User ID	RE1WE11-N02			**System ID**	2496853
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from Bwd 2016 NRC exam	**Difficulty**	
Technical Reference and Revision #	1BCA-1.1 Rev. 302		
Training Objective	T.CA2-05		
Previous Exam Use	None		

References Provided	None
K/A Justification	The question meets the K/A because it requires knowledge of the operating crew responsibilities of a continuous action during a loss of emergency coolant recirculation event.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring knowledge of EOP procedure.

K/A Reference(s)

P2.4.12	Safety Function	Tier	Group	RO Imp: 4.0	SRO Imp: 4.3
Knowledge of operating crew responsibilities during emergency and abnormal operations (CFR: 41.10 / 45.12)					
G.W.E11	Safety Function	Tier 1	Group	RO Imp:	SRO Imp:
Loss of Emergency Coolant Recirculation					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 18	ID: 2496857	Points: 1.00

- Unit 1 experienced a loss of heat sink and has entered 1BFR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.
- The crew is performing step 10, PREPARE TO ESTABLISH CONDENSATE BOOSTER FLOW, while monitoring for bleed and feed criteria.

Per the note prior to step 10, Bleed and feed should NOT be initiated due to low level in the SGs being depressurized, unless core exit temperatures are...

A. rising at any temperature.

B. above 557°F and rising.

C. above 700°F and rising.

D. above 1200°F and rising.

Answer	B

Answer Explanation

A - Plausible: Because several emergency procedure flow paths are directed by CETC trend direction but contain no specific temperature criteria.

B - Correct: Per 1BFR-H.1 note prior to step 10, "Bleed and feed should not be initiated due to low level in the SGs being depressurized, unless core exit temperatures are above 557°F and rising".

C - Plausible: Because CETC at 700°F or above is criteria for a core cooling orange path.

D - Plausible: Because CETC at 1200°F or above is criteria for a core cooling red path.

Question Information

Topic	Byron 2023 NRC Exam Q18				
User ID	RE1WE05-N03			System ID	2496857
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Modified from Byron 2022 NRC exam	Difficulty	N/A
Technical Reference and Revision #	1BFR-H.1, Rev. 304		
Training Objective	T.FR03-03		

Previous NRC Exams Use	None

References Provided	None
K/A Justification	Meets the K/A by requiring ability to interpret CETC during a loss of heat sink event.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's EOP procedure.

Original question:

- Unit 1 experienced a loss of heat sink and has entered 1BFR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.
- Containment pressure is 2.0 psig.
- The crew is performing step 11, TRY TO ESTABLISH CONDENSATE BOOSTER FLOW TO AT LEAST ONE SG.
- 1A, 1B and 1C SGs are being depressurized with ALL WR levels at 20%.
- 1D SG is isolated with WR level of 30%.
- CETC are 580°F and slowly DROPPING.

With the above conditions, should the crew IMMEDIATELY establish RCS Bleed and Feed, and why or why not?

A. NO, because CETC are dropping.

B. NO, because the 1D SG level is above the Feed and Bleed criteria.

C. YES, because CETC are above 557°F.

D. YES, because the 1A, 1B and 1C SG levels are below the Feed and Bleed criteria.

Answer: A

K/A Reference(s)

W.E05.EA2.08	Safety Function	Tier 1	Group	RO Imp: 3.7	SRO Imp: 3.9
Ability to determine and/or interpret the following as they apply to Loss of Secondary Heat Sink: (CFR: 41.10 / 43.5 / 45.13) CETs and/or subcooling					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 19	ID: 2496896	Points: 1.00

- Unit one is in a refueling outage.
- The Spent Fuel Pool transfer canal gate is CLOSED and the canal is drained.
- Annunciator 1-1-B7, SPENT FUEL PIT LEVEL HIGH LOW, alarms UNEXPECTEDLY.
- The crew enters 1BOA REFUEL-2, REFUEL CAVITY OR SPENT FUEL POOL LEVEL LOW.
- A Spent Fuel Pool leak was identified and ISOLATED.
- Spent Fuel Pool level dropped 2' below normal during the event.

Which TWO (either) methods should be used to re-fill the Spent Fuel Pool?
(assume all methods are available)

1. Make-up from the RWST
2. RMCS blended flow via a portable hose
3. PW aligned to the Spent Fuel Pool filter
4. Demineralized Water via a portable hose

A. 1 & 2

B. 1 & 3

C. 2 & 4

D. 3 & 4

Answer	A

Answer Explanation

A - Correct: Per 1BOA REFUEL -2, Caution prior to step 11, Non-borated water sources should only be used as a last resort, if fuel uncovery is imminent. Therefore, re-filling from the RWST or RMCS are the correct answers using borated sources.

B - Plausible: Because both the RWST and PW via the FC filter are hard piped connections to the Spent Fuel Pit.

C - Plausible: Because both the RMCS blender and Demineralized Water are via portable hoses to the Spent Fuel Pit. Portable hose filling of the Spent Fuel Pit is the normal method for filling when making up for evaporation losses.

D - Plausible: Because PW and Demineralized Water are both non-borated sources. Non-borated sources is the normal method for filling when making up for evaporation losses.

Question Information

Topic	Byron 2023 NRC Exam Q19				
User ID	RE10036-N01			**System ID**	2496896
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA REFUEL-2 Rev. 116		
Training Objective	T.OA30-03		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to prioritize significance of alarm.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's EOP procedure.

K/A Reference(s)

P2.4.45	Safety Function	Tier	Group	RO Imp: 4.1	SRO Imp: 4.3
Ability to prioritize and interpret the significance of each annunciator or alarm (CFR: 41.10 / 43.5 / 45.3 / 45.12)					
G.APE.036	Safety Function	Tier 1	Group	RO Imp:	SRO Imp:
Fuel Handling Incidents					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 20	ID: 2496908	Points: 1.00

- Unit 1 was at 100% power.
- Subsequently, a SG tube leak was identified on the 1D SG requiring a Unit 1 shutdown.
- The crew is performing 1BOA SEC-8, STEAM GENERATOR TUBE LEAK, step 16, COOLDOWN AND DEPRESSURIZE RCS TO STOP SG TUBE LEAK
- Currently, Unit 1 is in MODE 3 at 0% power.
  - ALL SG pressures are 1092 psig.
  - RCS pressure is 2235 psig.
  - RCS Tave is 557°F.

Which of the following is an acceptable combination of RCS Temperature and RCS Pressure to establish acceptable subcooling AND stop the primary to secondary leak per 1BOA SEC-8, step 16?

(REFERENCE PROVIDED)

	RCS Temperature	RCS Pressure
A.	537 °F	1212 psig
B.	527 °F	1072 psig
C.	500 °F	1050 psig
D.	507 °F	812 psig

Answer	B

Answer Explanation

A - Plausible: Because although the point is between the minimum and maximum RCS pressure lines in Figure 1BOA SEC-8-1, it is above the leaking SG pressure provided in the stem of the question. Step 16 of 1BOA SEC-8 requires RCS pressure to be decreased to less than the leaking SG pressure.

B - Correct: It's a point that falls between the minimum and maximum RCS pressure lines in Figure 1BOA SEC-8-1, and it is below the leaking SG pressure provided in the stem of the question.

C - Plausible: It's incorrect because it's a point that falls above the maximum RCS pressure line in Figure 1BOA SEC-8. It's plausible because it is at a lower pressure than the SG pressure provided in the stem. Step 16 of 1BOA SEC-8 requires RCS pressure to be decreased to less than the leaking SG pressure.

D - Plausible: It's incorrect because it's a point that falls below the minimum RCS pressure line in Figure 1BOA SEC-8. It's plausible because it is at a lower pressure than the SG pressure provided in the stem. Step 16 of 1BOA SEC-8 requires RCS pressure to be decreased to less than the leaking SG pressure.

Question Information

Topic	Byron 2023 NRC Exam Q20				
User ID	RE10037-N01			**System ID**	2496908
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	REFERENCE PROVIDED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from Bwd 2022 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA SEC-8 Rev. 112		
Training Objective	T.OA43-05		
Previous NRC Exams Use	None		

References Provided	1BOA SEC-8, Figure 1BOA SEC-8-1
K/A Justification	Meets the K/A by requiring ability monitor RCS during a SGTL event.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's AOP procedure.

K/A Reference(s)

APE.037.AA1.16	Safety Function	Tier 1	Group	RO Imp: 3.5	SRO Imp:
Ability to operate and/or monitor the following as they apply to a Steam Generator Tube Leak: (CFR: 41.7 / 41.10 / 45.5 / 45.6) RCS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 21	ID: 2496913	Points: 1.00

- Unit 1 reactor power is 65% and stable.
- The 1A CW Pump is OOS.
- Main Generator load is 800 MW and stable.
- 1C zone Main Condenser pressure is 1.6 inches HgA and stable.

The following occurs:

- 1100: The 1C CW Pump trips causing Main Condenser pressure to RISE at a constant rate of 0.2 inches HgA/minute.
- 1109: A ramp down at 10 MW/minute is commenced in an attempt to stabilize condenser pressure.

The RO is directed to monitor current parameters and keep the crew informed before reaching reactor trip criteria.

Which of the following is the closest approximate time that the unit will FIRST reach the unacceptable region of Figure 1BOA SEC-3-1?

(assume the Load DROP rate remains constant and the rate of Main Condenser pressure RISE remains constant throughout the event)

(reference provided)

A. 1119

B. 1124

C. 1127

D. 1132

Answer	C

Answer Explanation

A - Plausible: Because 1119 is approx. time pressure would reach the 5.5" horizontal line. (trip criteria at < 480 MW).

B - Plausible: Because 1124 is approx. time if plotted without accounting for first 9 min. of pressure rise with no ramp.

C - Correct: Using 1BOA SEC-3-1:
Plot initial conditions of 800 MW and 1.6" HgA
Pressure rises to 3.4" in first nine minutes.
Drop 10 MW and raise .2" HgA every subsequent minute and the plot will intercept the NOT ACCEPTABLE line at approx. 620 MW and 7.0" HgA pressure. 7.0" - 1.6" = 5.4" total pressure rise. 5.4"/0.2" per min = 27 min. after initiating event. 1100 + 27 = 1127.

D - Plausible: Because 1132 is approx. time pressure would reach the 8.0" horizontal line. (trip criteria at > 710 MW).

Question Information

Topic	Byron 2023 NRC Exam Q21				
User ID	RE10037-N03			**System ID**	2496913
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from Bwd 2016 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA SEC-3 Rev. 107		
Training Objective	T.OA38-03		
Previous NRC Exams Use	None on previous Byron exams		

References Provided	1BOA SEC-3, Figure 1BOA SEC-3-1
K/A Justification	Meets the K/A by requiring knowledge of operational implications of relationship of condenser vacuum to circulating water flow.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's AOP procedure.

K/A Reference(s)

APE.051.AK1.01	Safety Function	Tier 1	Group	RO Imp: 3.3	SRO Imp:
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of Condenser Vacuum: (CFR: 41.8 / 41.10 / 45.3) Relationship of condenser vacuum to circulating water, flow rate, and temperature					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 22	ID: 2496916	Points: 1.00

- A fire in the MCR has occurred.
- The crew is performing 1BOA PRI-5, CONTROL ROOM INACCESSIBILITY.
- An EO is verifying power to the 4 KV ESF busses just prior to initial activation of the RSP.

Per 1BOA PRI-5, which 4KV ESF load will the EO remove control power from, to DISABLE auto start capability?

A. CS pumps

B. CV pumps

C. RH pumps

D. SI pumps

Answer	A

Answer Explanation

A - Correct: Per 1BOA PRI-5, CONTROL ROOM INACCESSIBILITY, step 7 of Attachment B LOCAL OPERATOR ACTIONS, the control power fuses are pulled for ALL CS pumps to prevent automatic start of any CS pump. There are no controls for the CS pumps at the RSDP.

B - Plausible: Because the suction valves for the CV pumps are deenergized in 1BOA PRI-5 and manually aligned to the RWST.

C - Plausible: Because the RH system has to be locally aligned and started per 1BOA PRI-5, if the SM determines the unit will be taken to Mode 4 or 5.

D - Plausible: Because multiple SI equipment components are locally operated in 1BOA PRI-5, if SI actuated and SI termination is subsequently required.

Question Information

Topic	Byron 2023 NRC Exam Q22				
User ID	RE10068-N01			System ID	2496916
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	Difficulty	N/A
Technical Reference and Revision #	1BOA PRI-5 Rev. 111		
Training Objective	T.OA16-03		

Previous NRC Exams Use	None

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of relationship between control room evacuation and the ED system (4kv bus loads).
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's AOP procedure. The question is RO level because it requires knowledge of the overall mitigation strategy of 1BOA PRI-5.

K/A Reference(s)

APE.068.AK2.16	Safety Function	Tier 1	Group	RO Imp: 3.7	SRO Imp:
Knowledge of the relationship between Control Room Evacuation and the following systems or components: (CFR: 41.7 / 45.7) Electrical distribution system					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 23 ID: 2496963 Points: 1.00

- Unit 1 experienced a faulted 1A SG event at 100% power.
- Containment pressure rose above the criteria to enter 1BFR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE.
- The crew is currently performing 1BFR-Z.1.

The reason for 1BFR-Z.1, step 5, VERIFY MAIN STEAM ISOLATION is to STOP the steam flow...

A. from the AFFECTED SG.

B. from the UNAFFECTED SGs.

C. from the ALL SGs.

D. to LIMIT contamination of the secondary system.

Answer	**B**

Answer Explanation

A - Plausible: Because this is a protection provided by the MSIV on a break downstream of the MSIV on the affected SG (in combination with the unaffected SGs MSIVs.

B - Correct: Per BD-FR-Z.1, Main steam lines are isolated to either minimize the consequences of and/or terminate the mass and energy releases associated with a possible high energy secondary line break. The MSIV protection provided depends upon where the fault or tube rupture occurs. Per the MS lesson plan (derived from the Westinghouse system descriptions) there are 4 types of protection provided by main steam isolation depending upon the type of event:

1. Break in main steam line from S/G inside containment.
2. Break in main steam line outside containment, but upstream of the MSIV.
3. Break in steam header outside containment downstream of MSIV.
4. S/G tube rupture.

When single SG inside containment is faulted, the MSIV on the affected SG does not limit the steam flow from the affected SG, because the MSIV is located downstream of the fault. However, the unaffected SGs will also feed the fault via the common main steam lines or the common main steam header. Therefore, the protection provided in the question stem scenario (a single faulted SG inside containment) is preventing steam flow from the unaffected SGs

C - Plausible: Because this is a protection provided by the MSIVs on a break downstream of the MSIV on the affected SG.

D - Plausible: Because this is a protection provided by a MSIV on a ruptured SG.

Question Information

Topic	Byron 2023 NRC Exam Q23				
User ID	RE1WE14-N01			**System ID**	2496963
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BFR-Z.1 Rev. 300 BD-FR-Z.1 Rev. 300 N-BY-TQ-ILT-MS-23 ILT Main Steam lesson plan Rev. 12		
Training Objective	T.OA16-03		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of the reason for verifying MSLI has occurred for a high containment pressure event.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's EOP procedure background document.

K/A Reference(s)

W.E14.EK3.09	Safety Function	Tier	Group	RO Imp: 3.8	SRO Imp:
Knowledge of the reasons for the following responses and/or actions as they apply to High Containment Pressure: (CFR: 41.5 / 41.10 / 45.6 / 45.13) Ensuring that MSLI has occurred					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 24	ID: 2496973	Points: 1.00

- Unit 1 is at 100% power.
- Letdown flow is currently 75 gpm.
- The other two (2) letdown orifice isolation valves are CLOSED, but available.
- All other equipment is normally aligned for this power level.
- Elevated RCS coolant activity from I-131 is reported by Chemistry and the crew enters 1BOA PRI-4, HIGH REACTOR COOLANT ACTIVITY.

1. What is the mitigation strategy of 1BOA PRI-4?

2. Elevated I-131 activity is most likely indication of...?

A. 1. MAXIMIZE letdown
2. potential failed fuel.

B. 1. MAXIMIZE letdown
2. a rise in RCS piping corrosion.

C. 1. ISOLATE letdown
2. potential failed fuel.

D. 1. ISOLATE letdown
2. a rise in RCS piping corrosion.

Answer	A

Answer Explanation

A - Correct: The mitigation strategy of 1BOA PRI-4 includes maximizing letdown to clean up the RCS with a mixed bed demin. Elevated I-131 is a fission product that would be indicative of a fuel leak. Whereas elevated RCS piping corrosion would be indicated by increase of metallic elements such as conductivity, magnesium, aluminum, iron or nickel (all which are sampled for by chemistry).

B - Plausible: Because elevated RCS piping corrosion would be indicated by increase of metallic elements such as conductivity, magnesium, aluminum, iron or nickel (all which are sampled for by chemistry).

C - Plausible: Because elevated RCS activity would raise dose rates outside containment (especially during a LOCA outside containment) requiring letdown isolation. 1BOA PRI-1 has entry condition of rising Aux Bldg radiation and steps to isolate letdown.

D - Plausible: Because elevated RCS activity would raise dose rates outside containment (especially during a LOCA outside containment) requiring letdown isolation. 1BOA PRI-1 has entry condition of rising Aux Bldg radiation and steps to isolate letdown. Also, elevated RCS piping corrosion would be indicated by increase of metallic elements such as conductivity, magnesium, aluminum, iron or nickel (all which are sampled for by chemistry).

Question Information

Topic	Byron 2023 NRC Exam Q24				
User ID	RE10076-N01			**System ID**	2496973
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Modified from 2021 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA PRI-4 Rev.103		
Training Objective	T.OA15-03		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of the reason for response to and difference of fission product and corrosion product elevated activity is RCS.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information contained in the site's AOP procedure.

Original question:

- Unit 1 is at 100% power.
- Letdown flow is currently 75 gpm.
- The other two (2) letdown orifice isolation valves are CLOSED, but available.
- All other equipment is normally aligned for this power level.
- Elevated RCS coolant activity is reported by Chemistry and the crew enters 1BOA PRI-4, HIGH REACTOR COOLANT ACTIVITY and Tech Spec 3.4.16, RCS SPECIFIC ACTIVITY.

1. What is the mitigation strategy of 1BOA PRI-4?

2. Per the bases for LCO 3.4.16, RCS Specific Activity, RCS specific activity limits prevent exceeding dose acceptance criteria for which event(s)?

A. 1. MAXIMIZE letdown
2. SGTR or MSLB

B. 1. MAXIMIZE letdown
2. RCS LOCA outside containment

C. 1. ISOLATE letdown
2. SGTR or MSLB

D. 1. ISOLATE letdown
2. RCS LOCA outside containment

Answer: A

K/A Reference(s)

APE.076.AK3.01	Safety Function	Tier 1	Group	RO Imp: 3.1	SRO Imp:
Knowledge of the reasons for the following responses and/or actions as they apply to High Reactor Coolant Activity: (CFR: 41.5,41.10 / 45.6 / 45.13) RCS differentiating activity due to fission products and due to corrosion products from chemistry report					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 25	ID: 2499819	Points: 1.00

- Unit 1 is at 100% power.
- 1A SX pump is RUNNING.
- 1B SX pump is in STANDBY.

- The following annunciator alarms at 1PM06J:
  1-1-A2, CNMT LEAK DETECT FLOW HI

- An NSO reports that U-1 SX discharge pressure has dropped 5 psig since his MCB walkdown 15 minutes ago.
- 1PI-SX007, PP 1A DSCH PRESS is currently stable at 92 psig.

- No other alarms have come in and primary system parameters are all normal.
- The crew determines that a SX leak exists in U-1 containment and enter 1BOA PRI-7, ESSENTIAL SERVICE WATER MALFUNCTION.

With the above conditions, per 1BOA PRI-7, step 6.d RNO, the FIRST MCR action the crew will take to identify the leak location is...

A. shutdown 1A & C RCFCs.

B. close 1SX016A RCFC 1A & C SX INLET VLV.

C. shutdown 1B & D RCFCs.

D. close 1SX016B RCFC 1B & D SX INLET VLV.

Answer	C

Answer Explanation

A - Plausible: Because shutdown of the 1A and 1C RCFC would be the first action if the 1B SX pump was running.

B - Plausible: Because closing the 1SX016A would be performed, but not until the SX pumps are swapped and the 1A and 1C RCFCs are shutdown. Also the 1SX016A valve will not close (interlocked) with the 1A SX pump running.

C - Correct: The first actions in 1BOA PRI-7 (step 4 RNO) are to shut down the RCFC train for non-running (1B/D train) SX pump.

D - Plausible: Because closing the 1SX016B would be performed, but not until the 1B and 1D RCFCs are shutdown.

Question Information

Topic	Byron 2023 NRC Exam Q25				
User ID	RS20022-N02			System ID	2499819
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only

Question Type	Bank from 2009 Bwd NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA PRI-7, Rev. 111		
Training Objective	T.OA18-03		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to operate the service water system during a containment flooding event.
SRO-Only Justification	N/A
Additional Information	RO license level because it requires knowledge of overall mitigation strategy of AOP.

K/A Reference(s)

W.E15.EA1.04	Safety Function	Tier 1	Group	RO Imp: 3.5	SRO Imp:
Ability to operate and/or monitor the following as they apply to Containment Flooding: (CFR: 41.5 to 41.8 / 45.5 to 45.8) Service water					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 26	ID: 2496991	Points: 1.00

- Unit 1 experienced a reactor trip from full power.
- One minute later, a loss of offsite power occurred.

Currently (90 minutes after the loss of offsite power):

- The crew is implementing 1BEP ES-0.2, NATURAL CIRCULATION COOLDOWN.
- RCS pressure is 885 PSIG.
- ALL SG pressures are 763 PSIG.
- CETCs indicate 532°F.
- RCS cold leg temperatures indicate 515.5°F.

SATURATED conditions, if present, are indicated in the ___(1)___ .
Per BEP ES-0.2 the operators will ____(2)____.

A. (1) reactor vessel plenum
(2) actuate safety injection

B. (1) SG tubes
(2) raise the RCS cooldown to 100°F/hr

C. (1) reactor vessel plenum
(2) raise the RCS cooldown to 100°F/hr

D. (1) SG tubes
(2) actuate safety injection

Answer	A

Answer Explanation

A - Correct: Saturation temperature for 885 psig (900 psia) is 532°F. CETCs measure fluid temperature of the plenum, therefore the CETCs at 532°F is a direct indication of plenum saturation. 1BEP ES-0.2 OAS directs operators to actuate SI and go to 1BEP-0, REACTOR TRIP OR SAFETY INJECTION, if RCS subcooling is unacceptable. With the plenum at saturated conditions, there is no subcooling (i.e. unacceptable) so this criteria is met.

B - Plausible: Because A cold leg temperature of 515°F means the cold leg is at the saturation temperature for the SGs (515°F is the saturation temperature for 778 psia or 763 psig) and one of the indications used to determine that natural circulation is occurring (per Attachment B of 1BEP ES-0.1, REACTOR TRIP RESPONSE). Also, because changing the RCS cooldown rate would add subcooling if the RCS cooled down without depressurizing. However, the RCS also depressurizes during the cooldown which can lower subcooling.

C - Plausible: Also, because changing the RCS cooldown rate would add subcooling if the RCS cooled down without depressurizing. However, the RCS also depressurizes during the cooldown which can lower subcooling.

D - Plausible: Because A cold leg temperature of 515°F means the cold leg is at the saturation temperature for the SGs (515°F is the saturation temperature for 778 psia or 763 psig) and one of the indications used to determine that natural circulation is occurring (per Attachment B of 1BEP ES-0.1, REACTOR TRIP RESPONSE).

Question Information

Topic	Byron 2023 NRC Exam Q26				
User ID	RE10056-C02			**System ID**	2496991
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from 2013 Bwd NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Steam tables. 1BEP ES-0.2, NATURAL CIRCULATION COOLDOWN, Rev. 302.		
Training Objective	T.EP01-05		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring examinee to determine the correct indication for saturation in the plenum (based on CETC indication) and determine the correct operational response.
SRO-Only Justification	N/A
Additional Information	Meets Tier 1 by requiring information from EOP. RO license level because the question can be answered with knowledge of the procedure's overall mitigative strategy and overall sequence of events.

K/A Reference(s)

W.E09.EA2.06	Safety Function	Tier 1	Group	RO Imp: 3.7	SRO Imp: 4.0
Ability to determine and/or interpret the following as they apply to Natural Circulation Operations: (CFR: 41.10 / 43.5 / 45.13) Core exit temperatures and/or subcooling					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 27	ID: 2496992	Points: 1.00

Unit 1 is at 100% power, normal alignment.

The following then occurs:

- 1CC685, CC FROM RC PUMPS THERM BARR ISOL VLV, inadvertently CLOSES and will NOT re-open.
- The crew enters 1BOA PRI-6, COMPONENT COOLING MALFUNCTION.

How will the failure affect the RCPs?

A. Motor radial bearings and seal outlet temperatures will rise.

B. Seal outlet and pump lower radial bearing temperatures will rise.

C. Pump lower radial bearing and motor thrust bearing temperatures will rise.

D. Motor thrust bearing and motor radial bearings temperatures will rise.

Answer	B

Answer Explanation

A - Plausible: The Component Cooling Water supply to the RCP motor bearings and thermal barriers has a common supply line and separate return lines. 1CC9416 is the containment isolation valve on the return line from the motor bearings only. Therefore, closing 1CC9416 would isolate cooling water flow from the RCP motor bearings (radial and thrust). However, 1CC685 closing will not affect the motor bearings cooling flow.

B - Correct: The Component Cooling Water supply to the RCP motor bearings and thermal barriers has a common supply line and separate return lines. 1CC685 is the containment isolation valve on the return line from the thermal barriers only. Therefore, closing 1CC685 will isolate cooling water flow from the RCP thermal barriers only. RCP seal flow and pump lower radial bearings are adjacent to and protected from RCS temperatures by the thermal barrier.

C - Plausible: The Component Cooling Water supply to the RCP motor bearings and thermal barriers has a common supply line and separate return lines. 1CC9416 is the containment isolation valve on the return line from the motor bearings only. Therefore, closing 1CC9416 would isolate cooling water flow from the RCP motor bearings (radial and thrust). However, 1CC685 closing will not affect the motor bearings cooling flow.

D - Plausible: The Component Cooling Water supply to the RCP motor bearings and thermal barriers has a common supply line and separate return lines. 1CC9416 is the containment isolation valve on the return line from the motor bearings only. Therefore, closing 1CC9416 would isolate cooling water flow from the RCP motor bearings (radial and thrust). However, 1CC685 closing will not affect the motor bearings cooling flow.

Question Information

Topic	Byron 2023 NRC Exam Q27				
User ID	RE10056-C02			**System ID**	2496992
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from 2022 Bwd Cert exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA PRI-6 Rev. 114 BOP RC-1 Rev. 49 P&ID M-66 sht 1 Rev. C		
Training Objective	S.RC2-09-E		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring examinee to demonstrate the ability to monitor RCP cooling water supply and know how a malfunction will affect the RCPs.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.003.A4.08	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp:
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) RCP cooling water supplies					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 28	ID: 2496993	Points: 1.00

- Unit 1 is at 26% power.
- Steam Dumps are in the TAVE mode.
- Main Turbine load is 252 Mw.
- IMPULSE PRESS OUT and MW OUT are selected on DEH Operation Panel Graphic 5501.
- Rod Bank Select Switch is in MANUAL.
- The 1C RCP TRIPS.

A Reactor Trip DOES NOT occur, and NO operator actions are taken.

(1) Steady state to steady state, total RCS LOOP FLOW will lower...

(2) STEAM FLOW in the UNAFFECTED loops will...

A. (1) but remain greater than 3/4 full flow.
(2) LOWER.

B. (1) but remain greater than 3/4 full flow.
(2) RISE.

C. (1) to slightly less than 3/4 full flow.
(2) LOWER.

D. (1) to slightly less than 3/4 full flow.
(2) RISE.

Answer	B

Answer Explanation

A – Plausible: TOTAL RCS loop flow will remain greater than 3/4 full flow is correct. Unaffected loop steam flow will lower is incorrect. This would be the correct answer if the stem asked for unaffected loop steam pressure.

B – Correct: Rx power will remain constant due to constant steam demand. Reverse flow through the affected loop will raise flow in each of the unaffected loops. Lower RCS flow (less heating) in the affected loop will drop steam pressure and steam flow in the affected loop and subsequently raise steam flow in the unaffected loops also dropping steam pressure in them.

C – Plausible: TOTAL RCS loop flow lowering to less than 3/4 full flow is incorrect. Unaffected loops steam flow will drop is incorrect. This would be the correct answer if the stem asked for core flow vice loop flow and steam pressure of the unaffected loop.

D – Plausible: TOTAL RCS loop flow lowering to less than 3/4 full flow is incorrect. Unaffected loop steam flow will rise is correct. This would be the correct answer if the stem asked for core flow vice loop flow.

Question Information

Topic	Byron 2023 NRC Exam Q28				
User ID	RE10015-K1.04-04			**System ID**	2496993
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.3 Mechanical components and design features of the reactor primary system.

NRC Exams Only			
Question Type	Bank from 2019 Bwd NRC exam	**Difficulty**	
Technical Reference and Revision #	INPO 193006 - Fluids and Dynamics lesson plan Rev. 3.2 ILT RCP Lesson Plan N-BY-TQ-ILT-RC-13 Rev. 12		
Training Objective	S.RC2-07-B		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the KA because the examinee must evaluate the operational implications in an idle loop at power, its effect on RCS loop flow, and SG Pressure during a reactor coolant pump malfunction (trip).
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.003.K3.01	Safety Function	Tier 2	Group	RO Imp: 4.2	SRO Imp:
Knowledge of the effect that a loss or malfunction of the Reactor Coolant Pump System will have on the following systems or system parameters: (CFR: 41.7 / 45.6) RCS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 29	ID: 2496998	Points: 1.00

- Unit 1 at 100% power.
- RMCS is unavailable.
- At 1300, VCT level on both channels is 40% and lowering 1% per minute due to an un-isolable letdown line leak.

With NO operator action, what is the LATEST time the CV pump suction will swap to the RWST?

A. 1303

B. 1320

C. 1335

D. 1340

Answer	C

Answer Explanation

A - Plausible: Because the VCT auto makeup would normally occur at 37% (in 3 minutes) and without makeup capability, the only way to ensure VCT level does not lower more is to align the RWST. This is incorrect because the RWST swap will automatically occur at 5%.

B - Plausible: Because the VCT low level alarm will actuate at 20% (in 20 minutes). This would be an indication that make up is required and without makeup capability the only way to ensure VCT level does not lower more is to align the RWST. The automatic swap to the RWST won't occur until 5% which makes this choice incorrect.

C - Correct: At 1% per minute (19.3 gal/min), it will take 35 minutes for the VCT level to drop to 5% and swap to the RWST suction. At this time the VCT level will begin to rise due to partial letdown input.

D - Plausible: Because the VCT level would indicate 0 at this time. This is the minimum readable level and cannot drop any lower. However, the automatic swap to the RWST takes place at 5% to protect the CV pumps from gas binding making this answer incorrect.

Question Information

Topic	Byron 2023 NRC Exam Q29				
User ID	RS20003-N03			System ID	2496998
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.3 Mechanical components and design features of the reactor primary system.

NRC Exams Only			
Question Type	Bank from 2018 Bwd NRC exam	**Difficulty**	
Technical Reference and Revision #	BAR 1-9-A2 Rev. 11		
Training Objective	S.CV1-06-E		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the KA because it requires the examinee to know the level at which the suction will align to the RWST to prevent losing suction to the CV pumps from the VCT.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.004.K4.12	Safety Function	Tier 2	Group	RO Imp: 3.9	SRO Imp:
Knowledge of Chemical and Volume Control System design features and/or interlocks that provide for the following: (CFR: 41.6 / 41.7) Automatic action(s), that occur based on level of VCT					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 30	ID: 2497003	Points: 1.00

- Unit 2 is in MODE 5.
- The 2A RH pump is in standby and aligned for shutdown cooling.
- The 2B RH train is in shutdown cooling.

Subsequently:

- Breaker 2422, SAT 242-2 Feed to 4KV Bus 242, tripped due to a phase A overcurrent condition on Bus 242.

Given the conditions above, to restore shutdown cooling, one minute later the…

A. 2A RH pump can be manually started.

B. 2B RH pump can be manually started.

C. 2A AND 2B RH pumps have automatically started.

D. 2A RH pump automatically started and the 2B RH pump did NOT automatically start.

Answer	A

Answer Explanation

A - Correct: Bus 242 is locked out due to a phase A overcurrent, only Bus 241 is available to start the 2A RH pump. Since there is no SI in progress, the RH pumps do not receive an automatic start signal.

B - Plausible: If examinee realizes that the RH pumps will not automatically start and does not realize that Bus 242 is locked out.

C - Plausible: If examinee realizes Bus 242 is locked out and thinks that on a loss of power the RH pumps will sequence on, in this case only the 2A RH pump. The RH pumps get an automatic start signal when a SI signal is present.

D - Plausible: If the examinee does not realize Bus 242 is locked out and thinks that on a loss of power the RH pumps will sequence on.

Question Information

Topic	Byron 2023 NRC Exam Q30				
User ID	RS1005-N14-13			System ID	2497003
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	Bank from 2020 Bwd NRC exam	**Difficulty**	
Technical Reference and Revision #	6E-2-4001A Rev. T 6E-2-4030EF01 Rev. O, 6E-2-4030EF02 Rev. O		
Training Objective	S.RH1-11		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the KA because examinee must understand how the loss of a power supply affects the RH pumps.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.005.K2.01	Safety Function	Tier 2	Group	RO Imp: 4.1	SRO Imp:
Knowledge of electrical power supplies to the following: (CFR: 41.7) RHR pumps					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 31 ID: 2497015 Points: 1.00

- Unit 1 was at 100% power.
- An RCS LOCA and automatic SI occurred.
- 1BEP ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, is in progress.
- When realigning the ECCS system to cold leg recirc, 1SI8804B, "RH HX 1B TO SI PMPS ISOL VLV," would NOT open.
- The crew has completed the procedure up to step 10, CHECK RWST EMPTY.

When RWST level reaches 9%, what will be the status of the U1 SI pumps?

A. NEITHER pump will have a recirc suction source and BOTH MUST be shutdown.

B. ONLY the 1A pump will have a recirc suction source, and the 1B pump MUST be shutdown.

C. ONLY the 1B pump will have a recirc suction source, and the 1A pump MUST be shutdown.

D. BOTH SI pumps will have a recirc suction source and can remain running.

Answer	D

Answer Explanation

A - Plausible: Because the 1SI8804B is fed from the discharge from the 1B RH pump into the suction for the SI pumps. This is incorrect because the suction header forms a ring and suction will be maintained from the 1A RH pump.

B - Plausible: Because the 1A RH pump recirc discharge header feeds into the 1A SI pump side of the SI suction header. This is incorrect because the suction header forms a ring that includes connecting the suctions for the SI pumps.

C - Plausible: Because the 1B RH pump recirc discharge header feeds into the 1B SI pump side of the SI suction header. This is incorrect because the suction header forms a ring that includes connecting the suctions for the SI pumps.

D - Correct: The ECCS system is designed such that a single valve failure will not cause a loss of function. The RH pump supply to the CV and SI pumps suction has a crossover line between the CV and SI suction headers, such that either RH pump can supply suction to all the CV and SI pumps. 1BEP ES-1.3 accounts for a failure of either 1CV8804A or 1SI8804B to open, by only requiring one of the valves to be open when isolating the CV and SI pumps from the RWST (step 8a, open bullets).

Question Information

Topic	Byron 2023 NRC Exam Q31				
User ID	RS10006-N01			**System ID**	2497015
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	Bank from 2018 Bwd NRC exam	**Difficulty**	
Technical Reference and Revision #	ILT LP N -BY-TQ-ILT-ECCS-58 Rev. 13 1BEP ES-1.3 Rev. 305		
Training Objective	T.EP02-09-D		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the KA because examinee must predict the impact of a loss of flow path within the ECCS system and understand that the EOP will properly align the system considering the loss of flow path.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.006.A2.02	Safety Function	Tier 2	Group	RO Imp: 3.9	SRO Imp: 3.8
Ability to (a) predict the impacts of the following on the Emergency Core Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 45.5 / 45.3 / 45.4) Loss of flowpath					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 32	ID: 2497019	Points: 1.00

- Unit 1 is in Mode 2.
- Annunciator 1-12-D6, PZR SAF RLF DSCH TEMP HIGH alarms.
- The crew has determined that one pressurizer safety valve is leaking by.

With the above condition...

A. ONLY PRT pressure will rise to 100 psig, then the PRT rupture disk will rupture.

B. ONLY RCDT pressure will rise to 100 psig, then the RCDT relief valve will lift.

C. BOTH PRT AND RCDT pressure will rise to 100 psig when the PRT rupture disk and the RCDT relief valve will open.

D. BOTH PRT AND RCDT pressure will rise to 6 psig, then ONLY the PRT pressure will continue to rise to 100 psig when the PRT rupture disk will open.

Answer	A

Answer Explanation

A - Correct: The pressurizer safety valves exhaust to the PRT and the common drain header for both tanks is normally isolated between the tanks. Additionally, the common vent header is normally isolated. Therefore, only the PRT pressures will rise and the rupture disk will rupture at 100 psig.

B - Plausible: Because the PRT and RCDT have common vent piping to the waste gas vent header and both tanks have 100 psig relief mechanisms. However, 1RY469 valve is normally closed, isolating the PRT from the waste gas vent header.

C - Plausible: Because the PRT and RCDT have common vent piping to the waste gas vent header and both tanks have 100 psig relief mechanisms. However, 1RY469 valve is normally closed, isolating the PRT from the waste gas vent header.

D - Plausible: Because a leaking pressurizer safety valve would initially raise both PRT and RCDT pressures if 1RY469 was open (common waste gas vent header is connected to both tanks). At 6 psig, 1RY469 valve will receive an auto close signal, isolating the PRT from the waste gas vent header. PRT pressure will continue to rise until the PRT rupture disks open at 100 psig.

Question Information

Topic	Byron 2023 NRC Exam Q32				
User ID	RS10007-N02			**System ID**	2497019
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	**Difficulty**	
Technical Reference and Revision #	ILT RY LP N-BY-TQ-ILT-RY-14 Rev.16 P&IDs M-60 sheet 5 Rev. AO and 6 Rev. AG M-70 sheet 1 Rev. AF		
Training Objective	S.RY1-14		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the KA requires knowledge of the operation implications of a leaking pressurizer safety valve.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.007.K5.08	Safety Function	Tier 2	Group	RO Imp: 4.0	SRO Imp:
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Pressurizer Relief Tank/Quench Tank System: (CFR: 41.5 / 45.7) Recognition of leaking PORVs/code or safety valves					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 33	ID: 2497059	Points: 1.00

- A localized instrument air (IA) leak required isolating IA to 1CC182, CC SURGE TK PRI WTR MU VLV, and 1CC183, CC SURGE TK DEMIN WTR MU VLV.
- No other valves are affected.

(1) What will result from the IA loss?

(2) What action is required?

A. (1) BOTH 1CC182 AND 1CC183 will fail CLOSED.

(2) 1CC183 will need to be locally opened if CC Surge Tank makeup is required.

B. (1) 1CC182 will fail CLOSED and 1CC183 will fail OPEN.

(2) WM will need to be locally isolated to the CC Surge Tank to prevent overfill.

C. (1) 1CC182 will fail OPEN and 1CC183 will fail CLOSED.

(2) PW will need to be locally isolated to the CC Surge Tank to prevent overfill.

D. (1) BOTH 1CC182 AND 1CC183 will fail OPEN.

(2) PW AND WM will need to be locally isolated to the CC Surge Tank.

Answer	B

Answer Explanation

A - Plausible: Because 1CC182 fails closed. However, 1CC183 will fail open.

B - Correct: 1CC182 will fail CLOSED. 1CC183 will fail OPEN. The local isolation of the failed open air operated valve could be administratively controlled with a with an EST tag. If the crew chose to use OA procedural guidance for the CC system malfunction, 1BOA SEC-4, Table A, contains note 11 that warns the crew a continuous Demin Makeup to the CC Surge tank will occur on a loss of instrument air. Also, 1BOA ELEC-4 directs the local isolation of the failed open valve. Although the reason for the local isolation in 1BOA ELEC-4 is loss of power to the valve solenoids, a loss of control air to the valves would have the same affect.

C - Plausible: Plausible because it is opposite of the correct answer.

D - Plausible: Because 1CC182 fails open. However, 1CC183 will fail closed.

Question Information

Topic	Byron 2023 NRC Exam Q33				
User ID	RS10008-N05			**System ID**	2497059
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	Bank from 2021 Byron NRC exam	**Difficulty**	
Technical Reference and Revision #	M-66 Sht. 4A Rev. AV		
Training Objective	S.CC1-18		
Previous NRC Exams Use	2021 Byron NRC exam		

References Provided	None
K/A Justification	Question meets the K/A, requires examinee to predict the impact of loss of IA on a CC system AOV and use procedures to mitigate the malfunction.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.008.A2.05	Safety Function	Tier 2	Group	RO Imp: 3.2	SRO Imp: 3.3
Ability to (a) predict the impacts of the following on the Component Cooling Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) Effect of loss of instrument and control air on the position of air-operated CCW valves					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 34	ID: 2496574	Points: 1.00

- Unit 1 is at 100% power, stable conditions.
- 1PK-455A, MASTER PZR PRESS CONT, output is failed as is.
- 1PK-455A will NOT transfer to Manual nor Local control.

- A grid disturbance results in a rapid runback to 70% power.
- The US directs the RO to manually operate Pressurizer Pressure controls during the runback.

Current parameters are as follows:
- PZR Level 66%
- PZR Pressure 2280 PSIG

Under these conditions the NSO will …

A. turn OFF ALL heaters, and initiate spray flow to compensate for the initial insurge.

B. verify ON the backup heaters and initiate spray flow to compensate for the initial insurge.

C. verify ON the backup heaters and CLOSE both Pressurizer spray valves to compensate for the initial outsurge.

D. turn OFF ONLY the backup heaters, and fully OPEN both Pressurizer spray valves to compensate for the initial outsurge.

Answer	B

Answer Explanation

A - Plausible: Because pressurizer pressure is rising, turning off pressurizer heaters is plausible. However, the insurge of relatively colder RCS water can cause the pressurizer to lose saturation conditions. Therefore, heaters are turned on when an insurge causes level to rise > 5% above program level.

B - Correct: The rapid turbine runback results in RCS heatup, causing an insurge into the Pressurizer. The automatic design is that all backup heaters turn on (at a setting of 5% above program) to heat up the cool water that surges into the bottom of the pressurizer to maintain saturation conditions. At the same time, the insurge compresses the steam space causing Pressurizer pressure to rise, necessitating opening pressurizer sprays. BAR 1-12-C3 directs the operator to ensure automatic actions occur of ALL backup heaters ON.

C - Plausible: Because an outsurge would be caused by a rapidly cooling RCS. Eventually, Tave will drop due to control rods stepping in and possible boration of the RCS to account for the turbine runback. However, current conditions in the question stem indicate an insurge. Closing spray valves would be appropriate if pressurizer pressure was dropping from an outsurge.

D - Plausible: Because an outsurge would be caused by a rapidly cooling RCS. Eventually, Tave will drop due to control rods stepping in and possible boration of the RCS to account for the turbine runback. However, current conditions in the question stem indicate an insurge. Operation of pressurizer controls is the opposite of appropriate actions for an outsurge.

Question Information

Topic	Byron 2023 NRC Exam Q34				
User ID	RE10027-N03			**System ID**	2496574
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.3 Mechanical components and design features of the reactor primary system.

NRC Exams Only			
Question Type	Bank from 2022 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	BAR 1-12-C3, Rev. 4		
Training Objective	S.RY1-07-D		
Previous 2 Byron NRC Exams Use	2022 Byron NRC exam		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to predict changes in parameters associated with operation of the PZR pressure control system including PZR pressure (and steam temperature in saturated conditions) by reducing steam pressure to normal operating pressure.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.010.A1.11	Safety Function	Tier 2	Group	RO Imp: 3.3	SRO Imp:
Ability to predict and/or monitor changes in parameters associated with operation of the Pressurizer Pressure Control System, including: (CFR: 41.5 / 45.5) PZR steam temperature					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 35	ID: 2497067	Points: 1.00

- Unit 1 is in Mode 3 NOP/NOT.
- The crew is performing 1BOSR 0.5-2.RY.2, 1RY455A AND 1RY456 STROKE AND POSITION INDICATION TEST.
- All steps, prior to stroke of 1RY456 are complete.

When PZR PORV 1RY456 is stroked open, the RO will time the stroke and expect to see PRT pressure __(1)__ and PRT temperature __(2)__.

A. (1) rise
(2) rise

B. (1) NOT change
(2) rise

C. (1) rise
(2) NOT change

D. (1) NOT change
(2) NOT change

Answer	D

Answer Explanation

A - Plausible: Because PORV isolation valves have leaked by in the past and this would cause PRT pressure and temperature to rise. No indication of leak by was given in the stem.

B - Plausible: It is plausible that temperature would rise from steam trapped between the PORV and the PORV isolation valve causing the water in the PRT to absorb the heat of condensation and raise the PRT temperature without significant pressure rise. The volume of steam trapped would not be significant enough to raise PRT temperature.

C - Plausible: It is plausible that hot water trapped between the PORV and the isolation valve at high pressure would be released into the PRT and flash to steam causing a pressure rise but not be a significant enough mass to affect temperature. The volume of water trapped would not be significant enough to raise PRT pressure.

D - Correct: The pressurizer PORVs are isolated by their block valves before testing and the PRT sees no energy addition during PORV testing.

Question Information

Topic	Byron 2023 NRC Exam Q35				
User ID	RS20010-N04			System ID	2497067
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	Bank from 2018 Bwd NRC exam	**Difficulty**	
Technical Reference and Revision #	1BOSR 0.5-2.RY.2 Rev. 14		
Training Objective	S.RY1-11		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	This question meets the K/A because it requires the examinee to understand how PORV testing affects the PRT.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.010.A4.06	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp:
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Cycling PORVs, including PRT/quench tank parameters					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 36	ID: 2497116	Points: 1.00

- Unit 2 is at 100% power, all systems normally aligned.
- An automatic reactor trip signal is generated.
- Due to a malfunction, SSPS Train 2A Logic Cabinet does NOT process the Reactor Protection logic coincidence.
- SSPS Train 2B functions as designed.

Based on the above conditions and assuming NO operator actions are taken, how will the 2A Reactor Trip Breaker respond?

A. REMAINS CLOSED

B. OPENS with ONLY a UV trip signal

C. OPENS with ONLY a shunt trip signal

D. OPENS with BOTH UV AND shunt trip signals

Answer	A

Answer Explanation

A - Correct: SSPS logic circuits provide train specific trips to the reactor trip breakers and opposite train trip to the reactor trip Bypass breakers (via only the UV coil). Therefore, failure of 2A train to process the logic coincidence would prevent 2A RTB from opening. 2B train would provide a trip signal to the 2A bypass breaker, but in a normally aligned condition the 2A Bypass breaker would be racked out.

B - Plausible: Because Train B SSPS processing an automatic reactor trip signal will de-energize the UV coil on the 2A Bypass Breaker. However, the 2A reactor trip breaker will not receive a trip signal.

C - Plausible: Because a manual reactor trip switch or a manual SI switch from A train will actuate the shunt coil on the 2A reactor trip breaker. However, the question stems states that no operator actions were taken.

D - Plausible: Because the combination of distractors B and C could trip the 2A reactor trip breaker with both actuation devices. However, neither plausible actuations will occur.

Question Information

Topic	Byron 2023 NRC Exam Q36				
User ID	RS20012-N04			System ID	2497116
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only

Question Type	Modified from 2009 Bwd NRC exam	**Difficulty**	
Technical Reference and Revision #	ILT SSPS lesson plan I1-RP-XL-01 Rev. 9		
Training Objective	S.RP1-09-A		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	This question meets the K/A because it requires the examinee ability to monitor RPS trip logic processing of SSPS.
SRO-Only Justification	N/A
Additional Information	None

Original question:
Given:
- Unit 2 is at 100% power, all systems normally aligned.
- An automatic reactor trip signal is generated.

- Due to a malfunction, SSPS Train 2A Logic Cabinet does NOT process the Reactor Protection logic coincidence.

- SSPS Train 2B functions as designed.

Based on the above conditions and assuming NO operator actions are taken, which ONE of the following would occur?

A. 2A Reactor Trip Breaker REMAINS CLOSED.
2B Reactor Trip Breaker OPENS with ONLY UV trip signals.

B. 2A Reactor Trip Breaker OPENS with ONLY UV trip signal.
2B Reactor Trip Breaker OPENS with UV AND shunt trip signals.

C. 2A Reactor Trip Breaker REMAINS CLOSED.
2B Reactor Trip Breaker OPENS with UV AND shunt trip signals.

D. 2A Reactor Trip Breaker OPENS with ONLY shunt trip signal.
2B Reactor Trip Breaker OPENS with UV AND shunt trip signals.

Answer	**C**

K/A Reference(s)

SYS.012.A3.06	Safety Function	Tier 2	Group	RO Imp: 4.1	SRO Imp:
Ability to monitor automatic features of the Reactor Protection System, including: (CFR: 41.6 / 41.7 / 45.5) Trip logic					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 37	ID: 2497120	Points: 1.00

- Unit 1 is in MODE 3 coming out of a refueling outage.

- Reactor trip breakers are CLOSED.
- ALL shutdown and control rods are fully inserted.
- Maintenance is being performed on the N-31 Source Range drawer.
- N-31 Level Trip switch is in BYPASS.
- ALL other plant equipment is normally aligned for MODE 3.

A malfunction occurs and the following annunciators are LIT:

- 1-10-A1 SR S/D FLUX HIGH
- 1-10-B1 SR NON-OPERATE

The reactor trip breakers did NOT open.

Which of the following fuses OPENING would cause the above plant response?

A. N-31 Control Power fuses

B. N-31 Instrument Power fuses

C. N-32 Control Power fuses

D. N-32 Instrument Power fuses

Answer	B

Answer Explanation

A – Plausible: Common misconception of system operation. If unit power had been above P-6 and N-31 SR level trip blocked, this would not have caused a reactor trip.

B – Correct: When Instrument power fuses blow, the N-31 detector will de-energize. This will cause annunciator 1-10-A1 and 1-10-B1 bistables to trip and alarms to come in. With the Level Trip switch in bypass, Control Power will maintain 1 of 2 SSPS input bay relays energized and prevent the reactor trip breakers from opening, therefore annunciator 1-10-A8 will not alarm.

C – Plausible: Common misconception of system operation. If N-32 SR level trip had been bypassed, this would not have caused a reactor trip.

D – Plausible: Common misconception of system operation. If unit power had been above P-6 and N-32 SR level trip blocked, this would not have caused a reactor trip.

Question Information

Topic	Byron 2023 NRC Exam Q37				
User ID	RE20032-N03			**System ID**	2497120
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	Bank from 2020 Bwd NRC exam	**Difficulty**	
Technical Reference and Revision #	ILT LP SR and IR NIs N-BY-TQ-ILT-NI-31 Rev.7		
Training Objective	S.NI1-08-A		
Previous Exam Use	None		

References Provided	None
K/A Justification	Meets K/A by requiring ability to monitor RPS indications during an event with a channel in bypass.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.012.A4.03	Safety Function	Tier 2	Group	RO Imp: 3.8	SRO Imp:
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Channel blocks and bypasses					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 38	ID: 2497133	Points: 1.00

- Unit 1 is at 100% power.
- Subsequently a 1A main feedline break occurs inside containment.
- 1 minute later, the following conditions exist:
  - 1A SG NR level is 17%.
  - Containment pressure is 4.0 psig.
  - 1A SG pressure is 600 psig.
  - RCS pressure is 2000 psig.
  - RCS Tave is 545°F
  - All MSIVs AUTOMATICALLY closed.

Which of the following ESF signals caused the MSIVs to close?

A. Low SG level

B. Low steam line pressure

C. High containment pressure

D. LO-2 Tave Interlock P-12

Answer	B

Answer Explanation

A – Plausible: Because Low SG level at 17% will cause a RPS actuation and SG level is expected to lower, but it will not cause a MSI signal.

B – Correct: Low steam line pressure of 640 psig (above P-11) will cause a MSI.

C – Plausible: Because High containment pressure of 3.4 psig will cause a SI actuation, but the setpoint for MSI is 8.2 psig which has not been reached.

D – Plausible: Because the P-12 interlock (Tave <550°F) will close the Steam Dumps but not the MSIVs

Question Information

Topic	Byron 2023 NRC Exam Q38				
User ID	RS20013-N04			System ID	2497133
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	New for 2023 Byron NRC	Difficulty	

Technical Reference and Revision #	BAR 1-15-E8 Rev. 4
Training Objective	S.MS1-07-D
Previous Exam Use	None

References Provided	None
K/A Justification	Meets K/A by requiring knowledge of how a main feedline break will affect the ESFAS.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.013.K6.10	Safety Function	Tier 2	Group	RO Imp: 4.0	SRO Imp:
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Engineered Safety Features Actuation System: (CFR: 41.6 / 41.7 / 41.8 / 45.5 to 45.8) Feedline break					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 39 ID: 2497147 Points: 1.00

- Unit 1 is at 100% power.
- The 1A Containment Chiller TRIPS.
- The 1B Containment Chiller will NOT start.
- Containment pressure is POSITIVE 0.1 psig and RISING at 0.1 psig per hour.

Currently, Tech Spec 3.6.6, Containment Spray and Cooling Systems _(1)_ required to be entered.

At the current rate, the Containment Pressure Tech Spec UPPER limit will be reached in _(2)_ hours.

A. (1) IS
(2) 7

B. (1) IS
(2) 9

C. (1) is NOT
(2) 7

D. (1) is NOT
(2) 9

Answer	D

Answer Explanation

A – Plausible: Because the Containment chillers are part of the containment cooling system. However, they are not safety related even though they are cooled by a safety related system (SX). Additionally, 7 hours would be the time required to reach the alarm setpoint (+0.8 psig) of 0-31-D10, CNMT INTERNAL PRESS HIGH LOW. An examinee may recall the setpoint of 0-31-D10, and calculate the time to this value. This also correlates to the indicating band of 1PDI-VP236 turning to yellow.

B – Plausible: Because the Containment chillers are part of the containment cooling system. However, they are not safety related even though they are cooled by a safety related system (SX).

C – Plausible: Because 7 hours would be the time required to reach the alarm setpoint (+0.8 psig) of 0-31-D10, CNMT INTERNAL PRESS HIGH LOW. An examinee may recall the setpoint of 0-31-D10, and calculate the time to this value. This also correlates to the indicating band of 1PDI-VP236 turning to yellow.

D – Correct: Containment chillers are not safety related even though they are cooled by a safety related system (SX) and therefore TS 3.6.6 entry is not required. Per Technical Specification LCO 3.6.4, the upper Technical Specification Containment Pressure limit is 1.0 psig. Since initial pressure was positive 0.1 psig, and Containment pressure is rising at .1 psig/hr, the limit will be reached in 9 hours.

Question Information

Topic	Byron 2023 NRC Exam Q39				
User ID	RS10022-N05			**System ID**	2497147
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	
Technical Reference and Revision #	TS 3.6.6 Amendment 209 and 3.6.4 Amendment 171		
Training Objective	S.VP1-10		
Previous Exam Use	None		

References Provided	None
K/A Justification	Meets K/A by requiring knowledge of the affect that a malfunction of the containment cooling system will have on containment.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.022.K3.04	Safety Function	Tier 2	Group	RO Imp: 3.9	SRO Imp:
Knowledge of the effect that a loss or malfunction of the Containment Cooling System will have on the following systems or system parameters: (CFR: 41.7 / 45.6) CNT					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 40	ID: 2497163	Points: 1.00

Per 1BGP 100-1 “Plant Heatup”, the MINIMUM temperature, above which, exceeds Excore Neutron Detectors and associated cables extended exposure limits, and would require an immediate start of a Reactor Cavity Vent Fan is...

A. 120°F

B. 135°F

C. 150°F

D. 160°F

Answer	B

Answer Explanation

A – Plausible: Because 1BGP 100-1, Limitation and Action E.3.g. contains a 120°F limit of "No more than one RCP SHALL be running when the RCS temperature is ≤120°F and the PZR is solid."

B – Correct: Per 1BGP 100-1, Precaution D.5.a., "MAINTAIN at least 1 Reactor Cavity Vent Fan ON whenever RCS temperature exceeds 135°F to prevent exposing Excore Neutron Detectors and associated cables to temperatures > 135°F for extended periods.

C – Plausible: Because 1BGP 100-1, Precaution D.2.c. contains a 150°F limit of "IF Seal Injection flow is lost with RCS temperature above 150°F, Component Cooling through the RCP Thermal Barriers shall be maintained."

D – Plausible: Because 1BGP 100-1, Limitation and Action E.3.h. contains a 160°F limit of "For solid water, low pressure operation at least one Reactor Coolant Pump must be running prior to increasing RCS temperature above 160°F."

Question Information

Topic	Byron 2023 NRC Exam Q40				
User ID	RS10022-N06			**System ID**	2497163
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.3 Mechanical components and design features of the reactor primary system.

NRC Exams Only			
Question Type	Bank from Byron 2021 Cert exam	**Difficulty**	
Technical Reference and Revision #	1BGP 100-1 Rev. 72		
Training Objective	S.VP1-13		
Previous Exam Use	None		

References Provided	None
K/A Justification	Meets K/A by requiring knowledge of the operational implications of electrical cabling exposed to high temperatures and when to ensure a containment fan is in operation.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.022.K5.05	Safety Function	Tier 2	Group	RO Imp: 3.0	SRO Imp:
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Containment Cooling System: (CFR: 41.5 / 45.7) Effects on electrical insulation					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 41	ID: 2497170	Points: 1.00

Following maintenance on valve 1CS009A, PP 1A SUMP SUCT VLV, an RO is performing 1BOSR 0.5-2.CS-1-1, TRAIN A CONTAINMENT SPRAY SYSTEM VALVE EXERCISE TEST.

In order to meet the interlocks to open 1CS009A, which of the following valves must be in the CLOSED position?

A. 1CS001A, PP 1A RWST SUCT VLV

B. 1SI8811A, CNMT SUMP 1A ISOL VLV

C. 1SI8812A, PP 1A SUCT FROM RWST ISOL VLV

D. 1RH8701A, RC LOOP 1A TO RH PP 1A SUCT ISOL VLV

Answer	D

Answer Explanation

A – Plausible: Because 1CS009A must be closed prior to opening 1CS001A (ensures ECCS sump flow path to RWST is not cross-tied).

B – Plausible: Because 1SI8811A must be open prior to opening 1CS009A (ensures ECCS sump flow path to CS). Although the surveillance position for 1SI8811A is closed, an electrical jumper is installed to simulate the valve being open.

C – Plausible: Because 1SI8812A is on the same suction header BUT has no interlock with 1CS009A.

D – Correct: 1RH8701A is interlocked to be closed prior to opening 1CS009A (ensures RCS hot leg not cross tied to CS suction).

Question Information

Topic	Byron 2023 NRC Exam Q41				
User ID	RS10026-N05			System ID	2497170
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	Difficulty	
Technical Reference and Revision #	1BOSR 0.5-2.CS-1-1 Rev. 19		
Training Objective	S.CS1-08-A		

Previous Exam Use	None

References Provided	None
K/A Justification	Meets K/A by requiring knowledge of the physical connections between CS and RHR.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.026.K1.06	Safety Function	Tier 2	Group	RO Imp: 3.6	SRO Imp:
Knowledge of the physical connections and/or cause and effect relationships between the Containment Spray System and the following systems: (CFR: 41.2 to 41.9 / 45.7 / 45.8) RHRS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 42	ID: 2497175	Points: 1.00

During normal at power plant operations, why is hydrazine added to the plant secondary system?

A. pH control

B. dissolved 0_2 control

C. balance the Na/Cl molar ratio in the SGs

D. reduce the amount of iron that deposits in the SGs

Answer	B

Answer Explanation

A – Plausible: Because MPA (Methoxy propylamine) is added to the secondary for pH control.

B – Correct: Per ILT lesson plan CD/FW system and CY-AP-120-2000F-02, continuous hydrazine is added to the secondary for dissolved 0_2 control.

C – Plausible: Because Ammonium Chloride is added to the secondary to balance the Na/Cl molar ratio in the SGs.

D – Plausible: Because PAA is added to bond with previously deposited iron in the SGs to promote removal via CPs and blowdown system (iron transport mitigation).

Question Information

Topic	Byron 2023 NRC Exam Q42				
User ID	RS10039-N08			**System ID**	2497175
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	
Technical Reference and Revision #	ILT CD/FW lesson plan N-BY-TQ-ILT-CD-FW-25 Rev. 16		
Training Objective	S.CD1-014-B		
Previous Exam Use	None		

References Provided	None
K/A Justification	Meets K/A by requiring knowledge of the reasons for secondary chemistry controls.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

P2.1.34	Safety Function	Tier	Group	RO Imp: 2.7	SRO Imp: 3.5
Knowledge of RCS or balance-of-plant chemistry controls, including parameters measured and reasons for the control (CFR: 41.10 / 43.5 / 45.12)					
G.SYS.059	Safety Function	Tier 2	Group	RO Imp:	SRO Imp:
MFW Main Feedwater System					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 43	ID: 2497398	Points: 1.00

When the stream dumps are in STM PRESS mode, what primary signal does the ACTIVE steam dump controller use to control steam dump valve demand?

A. Tref programmer output

B. PT-505, turbine impulse pressure

C. PT-507, MS cross tie header pressure

D. 2nd highest of PT-516/526/536/546 SG pressures

Answer	D

Answer Explanation

A – Plausible: Because the Tref programmer output is the reference signal to the steam dump load reject controller.

B – Plausible: Because 2 out of 3 logic of PT-505/506 and median of MS-002A/B/C turbine impulse pressures is the input to the Tref programmer. Also, PT-505 is an input to the C-7 arming signal for the steam dumps.

C – Plausible: Because PT-507 is the backup input to the steam pressure controller, but not the primary input.

D – Correct: The Ovation modification project changed the steam pressure controller input from PT-507 to a validated signal of 2nd highest of PT-516/526/536/546 SG pressures. PT-507 is now a backup if the Steam Pressure validation circuit goes into alternate action or can be manually selected.

Question Information

Topic	Byron 2023 NRC Exam Q43				
User ID	RS20041-N02			System ID	2497398
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	Difficulty	
Technical Reference and Revision #	ILT Steam Dump lesson plan N-BY-TQ-ILT-DU-24 Rev. 11		
Training Objective	S.DU1-06		
Previous Exam Use	None		

References Provided	None
K/A Justification	Meets K/A by requiring knowledge of the cause and effect relationship between the MSS and SDS.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

SYS.039.K1.06	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp:
Knowledge of the physical connections and/or cause and effect relationships between the Main and Reheat Steam System and the following systems: (CFR: 41.2 to 41.9 / 45.7 / 45.8) SDS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 44 ID: 2497455 Points: 1.00

- Unit 1 is ramping up per 1BGP 100-3, POWER ASCENSION 5 PERCENT TO 100 PERCENT.
- The BOP is monitoring Automatic Feed Reg valve and Feed Reg Bypass valve control.

What is the approximate individual STEAM FLOW from a SG, when the FW Bypass Reg Valves (1FW510A/520A/530A and 540A) automatically FULLY close?

A. 5%

B. 15%

C. 30%

D. 50%

Answer	**C**

Answer Explanation

A – Plausible: Because when the Feed Reg Bypass valve is at approx. 50% open, the associated Main Feed Reg valve will get a signal to open to 5%.

B – Plausible: Because during Power Descension the High Power SG water level controller will automatically swap to the Low Power controller when Feed Flow for that individual steam generator is less than 15% (approx. 0.6 MLBM).

C – Correct: When the individual steam flow to a steam generator reaches approx. 30% (approx. 1.2 MLBM) the Feed Reg Bypass valve will get a closed signal transferring control solely to the Main Feed Reg valve.

D – Plausible: Because when the Feed Reg Bypass valve is at approx. 50% open, the associated Main Feed Reg valve will get a signal to open to 5%.

Question Information

Topic	Byron 2023 NRC Exam Q44				
User ID	RS10059-A1.03-45			**System ID**	2497455
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

NRC Exams Only			
Question Type	Bank from 2019 Bwd NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BGP 100-3, Rev. 107		
Training Objective	S.CD1-010-C		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requiring ability to monitor automatic functions of the MFW system controls during a power ascension.
SRO-Only Justification	Not applicable
Additional Information	

K/A Reference(s)

SYS.059.A4.03	Safety Function	Tier 2	Group	RO Imp: 3.6	SRO Imp:
Ability to manually operate and monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Feedwater control during power increase and decrease					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 45	ID: 2497458	Points: 1.00

- A Unit 1 RCS cooldown is in progress with the 1A AF pump maintaining stable level in all steam generators.
- Then instrument power is lost to the 1AF005A controller.

Assuming RCS heat input and SG steaming rate remain constant, 1AF005A, SG 1A FLOW CONT VLV ...

A. fails closed and 1A SG level will drop.

B. fails open and 1A SG level will rise.

C. fails as is and 1A SG level will remain the same.

D. is NOT affected and 1A SG level will remain the same.

Answer	A

Answer Explanation

A – Correct: If instrument power is lost to the 1AF005A-D valve controllers, zero flow is demanded and the valves throttle closed to achieve zero flow.

B – Plausible: Because the 1AF005A-D MCR controllers require instrument power to operate.

C – Plausible: Because other plant control valves fail as is on a loss of control power (e.g. the 1AF013A-D valves).

D – Plausible: Because the power loss was AC vs. DC (DC loss would have no affect).

Question Information

Topic	Byron 2023 NRC Exam Q45				
User ID	RS20061-N01			System ID	2497458
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

NRC Exams Only			
Question Type	Bank from 2009 Bwd NRC exam	Difficulty	N/A
Technical Reference and Revision #	ILT AF lesson plan N-BY-TQ-ILT-AF-26 Rev. 13		
Training Objective	S.AF1-15		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requiring knowledge of the effect that a AFW system malfunction will have on

	the SGS.
SRO-Only Justification	Not applicable
Additional Information	

K/A Reference(s)

SYS.061.K3.02	Safety Function	Tier 2	Group	RO Imp: 4.3	SRO Imp:
Knowledge of the effect that a loss or malfunction of the Auxiliary/Emergency Feedwater System will have on the following systems or system parameters: (CFR: 41.5, 41.7 / 45.6) S/G system					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 46 ID: 2497470 Points: 1.00

- Unit 1 was at 100% power normally aligned.
- Subsequently, a reactor trip and main generator trip occurred.

Concerning the following 6.9 kV Bus 156 feed breakers:
- 1561 UAT Feed Breaker to Bus 156
- 1562 SAT Feed Breaker to Bus 156

Which of the following is the indication status of the breakers MCR position indication and control switch targets?
(Assume NO operator actions.)

		Position light	C/S target
A.	ACB 1561	green	red
	ACB 1562	green	green
B.	ACB 1561	green	green
	ACB 1562	green	red
C.	ACB 1561	red	green
	ACB 1562	red	red
D.	ACB 1561	red	red
	ACB 1562	red	green

Answer	**D**

Answer Explanation

A – Plausible: Because this is how the indications would have looked prior to the reactor trip.

B – Plausible: Because this is how the indications would have looked if it was aligned to the SATs and did not ABT (i.e. Bus 158 and 159 feed breakers).

C – Plausible: Because this is how the indications would have looked if the reactor tripped from low power operations (before bus 156 was aligned to the UAT)

D – Correct: Bus 156 is normally fed from the UATs at 100% power. When a reactor trip occurs, the UATs are de-energized and Bus 156 will ABT to the SATs. Therefore, the UAT feed breaker (1561) will indicate open which is red (not normal for 100% power). The SAT feed breaker (1562) will indicate closed which is also red (not normal for 100% power). ACB 1561 C/S target will be red because the breaker was in normal after close (NAC) at 100% power. ACB 1562 C/S target will be green because the breaker was in normal after trip (NAT) at 100% power.

Question Information

Topic	Byron 2023 NRC Exam Q46				
User ID	RS20062-N01			**System ID**	2497470
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	ILT lesson plan: AC Electrical Distribution N-BY-TQ-ILT-AP-4 Rev. 14 INPO fundamental LP: Breakers, Relays and Disconnects Rev. 3.1		
Training Objective	S.AP1-10-F		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requiring ability to recognize an auto bus transfer.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.062.A3.07	Safety Function	Tier 2	Group	RO Imp: 3.6	SRO Imp:
Ability to monitor automatic operation of the AC Electrical Distribution, including: (CFR: 41.7 / 45.5) Automatic bus transfer					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 47	ID: 2497475	Points: 1.00

- Both Units are at 100% power.
- 345 kV Bus 10 faults.

With the above condition, which Byron switchyard line will de-energize?

A. L 0621 to Cherry Valley

B. L 0622 to Cherry Valley

C. L 0624 to Wempleton

D. L 0627 to Lee County

Answer	C

Answer Explanation

A - Plausible: Because this is an offsite power source to the Byron switchyard. However, L0621 is connected to Byron switchyard bus 5.

B - Plausible: Because this is an offsite power source to the Byron switchyard. However, L0622 is connected to Byron switchyard bus 12.

C - Correct: A bus 10 fault will initiate a trip signal to 345 kV BT 10-14 and 10-11, and a direct transfer trip signal to the L0624 associated breakers at Wempleton switchyard. This will de-energize L0624.

D - Plausible: Because this is an offsite power source to the Byron switchyard. However, L0627 is connected to Byron switchyard bus 3.

Question Information

Topic	Byron 2023 NRC Exam Q47				
User ID	RS10062-C02			**System ID**	2497475
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	Modified from Byron 2022 NRC Exam	**Difficulty**	N/A
Technical Reference and Revision #	ILT lesson plan: AC Electrical Distribution N-BY-TQ-ILT-AP-4 Rev. 14		
Training Objective	S.AP1-04		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requiring knowledge of cause and effect relationship on the AC electrical distribution system of a faulted bus.
SRO-Only Justification	Not applicable
Additional Information	None

Original Question:
- Unit 1 tripped from 100% power.
- 345 kV BT 3-4 FAILED to open.

With the above condition, which Byron switchyard line will de-energize?

A. L 0621 to Cherry Valley

B. L 0622 to Cherry Valley

C. L 0624 to Wempleton

D. L 0627 to Lee County

Answer: D

K/A Reference(s)

SYS.062.K5.05	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp:
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AC Electrical Distribution System: (CFR: 41.5 / 45.7) Fault on a bus					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 48 ID: 2497492 Points: 1.00

- Both Units are at 100% power.
- Unit 1 has indication of a DC Bus 111 ground.
- Troubleshooting has determined the ground to be on DC Bus 111 Battery.
- The crew has just crosstied DC Bus 111 to 211 per BOP DC-7, 125V DC ESF BUS CROSS-TIE/RESTORATION and ISOLATED Battery and Battery Charger 111.

A Tech Spec entry for 3.8.4 DC Sources-Operating is required for...

A. Unit 1 IMMEDIATELY ONLY; NOT required for Unit 2.

B. BOTH Units IMMEDIATELY.

C. Unit 1 IMMEDIATELY and for Unit 2 AFTER an additional 2 hours.

D. Unit 1 IMMEDIATELY and will ONLY be required for Unit 2 if completion times are exceeded.

Answer	B

Answer Explanation

A – Plausible: This is plausible because the inoperable component is on Unit 1 and other Tech Spec functions, such as SX, supply both units and only 1 unit enters Tech Spec for their inoperable pump. It can incorrectly be viewed as the same for the battery charger supply to the DC bus.

B – Correct: Tech Spec 3.8.4 is not met on both units, Unit 1 because the battery charger is required to meet the LCO, Unit 2 because the DC bus is required to be independent of the opposite unit to meet the LCO.

C – Plausible: The inoperability of the DC bus on Unit 2 could be delayed by 2 hours if condition A was applied to the bus instead of the specific cross tie condition. This would make this choice plausible.

D – Plausible: This is plausible because Unit 1 is the unit with the inoperability and the LCO allows 2 hours to cross tie to the other unit's battery charger.

Question Information

Topic	Byron 2023 NRC Exam Q48				
User ID	RS20063-N05			**System ID**	2497492
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

NRC Exams Only			
Question Type	Bank from Bwd 2018 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Tech Spec 3.8.4 Amendment 212		
Training Objective	S.DC1-06		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requiring effect on the DC electrical system of a battery malfunction.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.063.K6.06	Safety Function	Tier 2	Group	RO Imp: 4.0	SRO Imp:
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the DC Electrical Distribution System: (CFR: 41.7 / 45.7) Battery					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 49	ID: 2497509	Points: 1.00

- Unit 1 is at full power, normal alignment with the following exception:
- 1BOSR 8.1.2-1, 1A DIESEL GENERATOR OPERABILITY SURVEILLANCE, is in progress.
- The DG 1A ACB 1413 Auto Reclose Circuit Arm Selector Switch is in SURV TEST.
- The 1A DG is running and supplying bus 141 in parallel with the SATs.

The following then occurs:

A Loss Of Offsite Power (LOOP) occurs due to a fault in SAT 142-1 actuating 86ST11A, SAT 142-1 Lockout Relay.

With the above conditions, breaker ACB 1413, DG 1A FEED TO 4KV BUS 141, will _____(1)_____.

Subsequently, the crew is crosstying bus 141 to 241 per 1BOA ELEC-4, LOSS OF OFFSITE POWER.

When crosstying bus 141 to 241, the DG 1A EMER MODE SPEED/VOLT C/S must be in _____(2)_____.

A. (1) REMAIN closed
(2) AUTO

B. (1) REMAIN closed
(2) MAN EMERG MODE

C. (1) TRIP open and RE-CLOSE
(2) AUTO

D. (1) TRIP open and RE-CLOSE
(2) MAN EMERG MODE

Answer	B

Answer Explanation

A – Plausible: Because AUTO is the normal position of the EMERG MODE SPEED VOLT CONT switch. However, if the EMERG MODE SPEED VOLT CONT switch was in AUTO, the DG would either pick up as much load as possible or drop all load, depending on grid frequency. Neither situation is acceptable.

B – Correct: When a DG is initially paralleled to the SATs, it is operating in droop mode to control DG load. A SAT fault will trip breaker 1412 (SAT feed to bus 141) open but not 1413 because it is only a trip of 1412 vs. a lockout of 1412 (which would trip 1413 open). Therefore 1413 remains closed through the event. Additionally, the trip of 1412 affects the DG control circuit by energizing relay 3IMX. This relay (when energized) places the DG in isochronous mode. Selecting MAN EMERG MODE enables speed and voltage control. The DG remains in isochronous until the SAT or Reserve feed breaker is closed, when the DG will return to droop. If the EMERG MODE SPEED VOLT CONT switch was in AUTO, the DG would either pick up as much load as possible or drop all load, depending on grid frequency. Neither situation is acceptable. 1BOA ELEC-4 directs the DG to be placed in MAN EMERG MODE prior to crosstying of bus 141 to 241.

C – Plausible: Because, if the DG started on a SI signal (vs. LOOP) ACB 1413 would trip. Also, because AUTO is the normal position of the EMERG MODE SPEED VOLT CONT switch. However, if the EMERG MODE SPEED VOLT CONT switch was in AUTO, the DG would either pick up as much load as possible or drop all load, depending on grid frequency. Neither situation is acceptable.

D – Plausible: Because, if the DG started on a SI signal (vs. LOOP) ACB 1413 would trip.

Question Information

Topic	Byron 2023 NRC Exam Q49				
User ID	RS20064-N04			**System ID**	2497509
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.8 Components, capacity, and functions of emergency systems.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	ILT Lesson plan AC Distribution N-BY-TQ-ILT-AP-4 Rev. 14 ILT Lesson plan Emergency Diesel Generators N-BY-TQ-ILT-DG-9 Rev. 11a 1BOA ELEC-4 Rev. 113		
Training Objective	S.DG1-03-E		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requiring ability to predict the impacts of an ESFAS actuation on the EDG and use procedures to control the consequences.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.064.A2.29	Safety Function	Tier 2	Group	RO Imp: 4.3	SRO Imp: 4.3
Ability to (a) predict the impacts of the following on the Emergency Diesel Generators and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) ESFAS actuation					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 50	ID: 2497523	Points: 1.00

- Unit 1 is at 100% power.
- A loss of 4KV Bus 142 has occurred.
- The 1B DG did NOT start.
- An EO is locally performing 1BOA ELEC-3, Attachment D, LOCAL START OF 1B DG.
- When the DC Control Power Available lights are checked, they are noted to be NOT LIT.

Which of the following DC buses will the EO check the associated breaker closed for each indicating light on the 1B DG local control panel?

	DC POWER ON/BUS # 1 Light	DC POWER ON/BUS # 2 Light
A.	Bus 111	Bus 112
B.	Bus 112	Bus 111
C.	Bus 112	Bus 112
D.	Bus 112	Bus 114

Answer	C

Answer Explanation

A – Plausible: Because this would assume that circuit #1 is fed from bus 111 and circuit #2 is fed from bus 112 (misconception that the indicating light nomenclature is associated with respective DC ESF buses).

B – Plausible: Because this would assume that circuit #1 is fed from bus 112 and circuit #2 is fed from bus 111 (misconception that the buses are redundant with a "normal" feed from division 12 and a "reserve feed from div.11. RCP trip circuits have a similar control power arrangement as this distractor).

C – Correct: Each DG has two control power circuits which supply different multiple functions of the DG auxiliaries. These circuits are NOT redundant (common misconception) and both are supplied from the DC ESF bus in the same division as the DG. For the 1B DG, each circuit is fed from a separate supply breaker on DC bus 112.

D – Plausible: Because this would assume that circuit #1 is fed from bus 112 and circuit #2 is fed from bus 114 (misconception that there is both ESF and NON-ESF functions fed by the circuits). This is plausible since 1BOA ELEC-3 allows the DG to be started with circuit #2 de-energized.

Question Information

Topic	Byron 2023 NRC Exam Q50				
User ID	RS20064-N03			**System ID**	2497523
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	Bank from 2016 Bwd NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	6E-1-4030DG51 Rev. AP 6E-1-4030DG52 Rev. AG 1BOA ELEC-3, Attachment D, step 4 Rev. 109		
Training Objective	S.DG1-03-D		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requiring ability to validate indications at the DG local control panel by checking the associated power supplies.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

P2.1.45	Safety Function	Tier	Group	RO Imp: 4.3	SRO Imp: 4.3
Ability to identify and interpret diverse indications to validate the response of another indication. (CFR: 41.7 / 43.5 / 45.4)					
G.SYS.064	Safety Function	Tier 2	Group	RO Imp:	SRO Imp:
EDG Emergency Diesel Generators					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 51	ID: 2497535	Points: 1.00

- Unit 1 is at 100% power, normal alignment.
- An alarm on the RMS indicates 1PR08J, SG Blowdown from Green to RED.

With the above condition, the...

A. 1SD02PA/B, Blowdown Condenser Hotwell Pumps, TRIP.

B. 1SD007, SG Blowdown Condenser Inlet Valve, CLOSES.

C. 1PS179A-D, SG Blowdown Sample Isolation Valves, CLOSE.

D. 0WX119A, CST Inlet Header Isolation from Blowdown Demin 0A Valve, CLOSES.

Answer	C

Answer Explanation

A – Plausible: Because this is an interlock for low condenser level or high demin inlet pressure.

B – Plausible: Because this is an interlock for high blowdown condenser pressure.

C – Correct: Per BAR RMS-1-1PR08J, this is an interlock for high radiation in the 1PR08J, SG Blowdown Rad Monitor.

D – Plausible: Because a high radiation signal from 0PR016J will swap SG blowdown flow from the main condenser to the SG blowdown monitor tanks by closing 0WX119A and opening 0WX058A.

Question Information

Topic	Byron 2023 NRC Exam Q51				
User ID	RS10073-N04			System ID	2497535
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

NRC Exams Only			
Question Type	Modified from 2016 Bwd NRC exam	Difficulty	N/A
Technical Reference and Revision #	BAR RMS-1-1PR08J Rev. 0		
Training Objective	S.AR1-04-B-3		

Previous NRC Exam Use	None

References Provided	None
K/A Justification	This question meets the K/A by requiring knowledge of interlock for system actuation on PRM.
SRO-Only Justification	Not applicable
Additional Information	None

Original question:
Given:

- Unit 1 is at 100% power, normal alignment.
- An alarm on the RM-11 indicates 0PR16J, SG Blowdown After Filter Outlet Rad Monitor has turned from Green to RED.

With the above condition, the...

A. 1SD02PA/B, Blowdown Condenser Hotwell Pumps, TRIP.

B. 1SD007, SG Blowdown Condenser Inlet Valve, CLOSES.

C. 1PS179A-D, SG Blowdown Sample Isolation Valves, CLOSE.

D. 0WX119A, CST Inlet Header Isolation from Blowdown Demin 0A Valve, CLOSES.

Answer	**D**

K/A Reference(s)

SYS.073.K4.02	Safety Function	Tier 2	Group	RO Imp: 3.7	SRO Imp:
Knowledge of Process Radiation Monitoring System design features and/or interlocks that provide for the following: (CFR: 41.7) System actuations based on PRM signals					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 52	ID: 2497550	Points: 1.00

- Both units are at 100% power.
- Bus 242 faults and de-energizes.

Which pair of SX Cooling Tower fans will NOT be available?

A. 0A and 0B fans on the 0A Cooling Tower

B. 0C and 0D fans on the 0A Cooling Tower

C. 0E and 0F fans on the 0B Cooling Tower

D. 0G and 0H fans on the 0B Cooling Tower

Answer	D

Answer Explanation

A – Plausible: Because the 0A and 0B fans are on the 0A SX cooling tower and bus 241 (also from Unit 2) powers a pair of fans on the 0A SX cooling tower.

B – Plausible: Because the 0C and 0D fans are on the 0A SX cooling tower and bus 241 (also from Unit 2) powers a pair of fans on the 0A SX cooling tower.

C – Plausible: Because the 0E and 0F fans are on the 0B SX cooling tower (first two 0B cooling tower fans alphabetically) and a common misconception is the 0B cooling tower fans are all power from Unit 2.

D – Correct: The 0G and 0H SX cooling tower fans are powered from 480v bus 232Z which is fed from 4kV bus 242.

Question Information

Topic	Byron 2023 NRC Exam Q52				
User ID	RS10076-N05			System ID	2497550
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	Difficulty	N/A
Technical Reference and Revision #	BOP SX-E3 Rev. 8		
Training Objective	S.SX1-15		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requiring knowledge of electrical power supplies to SX cooling tower fans.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.076.K2.07	Safety Function	Tier 2	Group	RO Imp: 2.5	SRO Imp:
Knowledge of electrical power supplies to the following: (CFR: 41.7) Cooling tower fans					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 53 ID: 2497603 Points: 1.00

- Unit 1 is at 100% power.
- An inadvertent phase A signal actuates on 1A train ONLY.

How will the 1IA065 and 1IA066 Containment Isolation Valves react to this condition?
(assume NO operator action)

A. 1IA065 will CLOSE, but 1IA066 will NOT close.

B. 1IA066 will CLOSE, but 1IA065 will NOT close.

C. 1IA066 will close FIRST, then 1IA065 will close when it loses control air.

D. 1IA065 will close FIRST, then 1IA066 will close when it loses control air.

Answer	**D**

Answer Explanation

A – Plausible: Because 1IA065 is the only valve to receive the phase A signal.

B – Plausible: Because on several containment penetrations the outside containment isolation valve equipment ID number is numerically larger than the inside isolation valve equipment ID number (opposite configuration of the IA penetration). Examples are VQ, RF, SI and FC systems.

C – Plausible: Because this is the opposite response from the correct answer.

D – Correct: Train A Phase A signal will only directly close the 1IA065 valve. Because the 1IA066 valve (inside containment) is downstream of the 1IA065, 1IA066 will lose control air when the containment instrument air system bleeds down. Then the 1IA066 will fail closed.

Question Information

Topic	Byron 2023 NRC Exam Q53				
User ID	RS10078-N06			**System ID**	2497603
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	P&ID M-55 sht. 4 Rev. T		
Training Objective	S.SA1-08		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requiring ability to monitor instrument air containment isolation valves.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.078.A3.05	Safety Function	Tier 2	Group	RO Imp: 3.3	SRO Imp:
Ability to monitor automatic features of the Instrument Air System, including: (CFR: 41.7 / 45.5) Isolation of instrument air to containment					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 54	ID: 2497622	Points: 1.00

- Unit 2 is in Mode 1.
- The 2A, 2B, and 2D RCFC's are operating in high speed.
- The 2C RCFC is in standby.

The following indications are observed on the Unit 2 RCFC temperatures:
- 2A RCFC Inlet Temperature 119° F.
- 2B RCFC Inlet Temperature 118° F.
- 2C RCFC Inlet Temperature 127° F.
- 2D RCFC Inlet Temperature 121° F.

Per the Tech Spec 3.6.5, Containment Air Temperature...

A. the action requirement must be applied because the average of ALL the RCFC temperatures exceeds the LCO upper limit.

B. the action requirement must be applied because ONE of the OPERATING RCFC's temperatures is above the LCO upper limit.

C. NO action is necessary because the average temperature of ALL OPERATING RCFC's is below the LCO upper limit.

D. NO action is necessary because ALL the individual RCFC temperatures are within the appropriate LCO limit(s).

Answer	C

Answer Explanation

A – Plausible: Because the shut down RCFC is not calculated into the average.

B – Plausible: Because only one RCFC over the limit does not require TS entry.

C – Correct: 2BOSR 0.1-1,2,3, Modes 1,2,3 Shiftly Daily Operating Surv. step F.8 describes the method for calculating containment temperature for tech spec limit comparison. This method is to calculate the average of the inlet temperatures on the running RCFCs. Tech Spec 3.6.5 requires the average air temperature to be $\leq$ 120° F in modes 1, 2, 3 and 4.

D – Plausible: Because 2D RCFC is above the limit.

Question Information

Topic	Byron 2023 NRC Exam Q54				
User ID	RS10103-N02			System ID	2497622
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

NRC Exams Only

Question Type	Bank from Bwd 2011 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	2BOSR 0.1-1,2,3 Rev.94 Tech Spec 3.6.5 Amendment 171		
Training Objective	S.VP1-10		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requires examinee ability to monitor system parameters to prevent exceeding design limits (Technical Specifications).
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.103.A1.01	Safety Function	Tier 2	Group	RO Imp: 3.9	SRO Imp:
Ability to predict and/or monitor changes in parameters associated with operation of the Containment System, including: (CFR: 41.5 / 45.5) Containment pressure, temperature, and/or humidity					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 55	ID: 2497966	Points: 1.00

- Unit 1 is ramping down from full power.
- Rod Control is in auto.

At approximately 85% power:
- DRPI indication for rod D-4, in CBD is 192 steps.
- All other rods in CBD indicate 176 steps on their group step counter.

The ramp is stopped and rod control is placed in Manual.

The DRPI indication can be verified by QPTR being __(1)__ and delta I being __(2)__ in the affected quadrant.

A. (1) lower
(2) lower

B. (1) lower
(2) higher

C. (1) higher
(2) lower

D. (1) higher
(2) higher

Answer	D

Answer Explanation

A – Plausible: Because the majority of the rods are inserted and the flux profile will be lower than the full power profile.

B – Plausible: Because the majority of the rods are inserted more and the flux profile will be lower than the full power profile and the stuck rod will cause AFD to be higher in the affected quadrant.

C – Plausible: Because QPTR will be higher in the quadrant with the stuck rod.

D – Correct: With the rest of the control bank inserted as much as 20 steps lower in the core (considering maximum DRPI inaccuracy of ± 4 steps), flux will be higher in the quadrant the stuck rod is in. Therefore, delta I and QPTR will be higher in the affected quadrant.

Question Information

Topic	Byron 2023 NRC Exam Q55				
User ID	RE10005-N01			System ID	2497966
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.6 Design, components, and functions of reactivity control mechanisms and instrumentation.

NRC Exams Only			
Question Type	Bank from Bwd 2018 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Byron PDMS ILT lesson plan I1-CX-XL-02 Rev. 4		
Training Objective	S.CX8-01		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requires examinee knowledge of operational implications of RPIS vs. demand position.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.014.K5.02	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp:
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Rod Position Indication System: (CFR: 41.6 / 41.7 / 45.7) RPIS independent of demand position					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 56 ID: 2497974 Points: 1.00

Unit 1 is at 100% power.

Which of the following (if any) would cause an error from normal indication on the SPDS Subcooling display?

A. A CETC indicating 200°F LOWER than actual temperature ONLY.

B. A CETC indicating 200°F HIGHER than actual temperature ONLY.

C. BOTH, a CETC indicating 200°F HIGHER than actual temperature OR 200°F LOWER than actual temperature.

D. NEITHER a CETC indicating 200°F HIGHER than actual temperature NOR 200°F LOWER than actual temperature.

Answer	B

Answer Explanation

A – Plausible: Because the malfunctioning CETC is not accurate. However, because the CETC is reading abnormally low, it would not be used in the average of the 10 highest calculation input to the SPDS display.

B – Correct: The SPDS display for subcooling input from CETC is the average of 10 highest CETC temperatures. Therefore, a CETC indication 200°F higher than actual temperature would cause the CETC input to be 20°F higher than actual and reduce the subcooling display by 20°F.

C – Plausible: Because in both malfunctions a CETC is reading inaccurately.

D – Plausible: Because a CETC reading 200°F low would not be used in the average of the 10 highest calculation input to the SPDS display. Additionally, a CETC reading high could be manually removed from the average of the 10 highest calculation. A common misconception is that a failed CETC would be automatically removed from averaging circuits, because ovation will automatically remove failed temperature inputs to other calculated control circuits based upon deviation from other channels.

Question Information

Topic	Byron 2023 NRC Exam Q56				
User ID	RS20017-N03			**System ID**	2497974
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.6 Design, components, and functions of reactivity control mechanisms and instrumentation.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Byron Plant Computer ILT lesson plan N-BY-TQ-ILT-CX-56 Rev.12		
Training Objective	S.CX1-05		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requires examinee knowledge of the effect that a malfunctioning CETC will have on the SPDS display.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.017.K3.03	Safety Function	Tier 2	Group	RO Imp: 3.6	SRO Imp:
Knowledge of the effect that a loss or malfunction of the In-Core Temperature Monitor System will have on the following systems or system parameters: (CFR: 41.7 / 45.6) SPDS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 57	ID: 2498002	Points: 1.00

- Unit 1 is at 100% power.
- A containment mini-purge is in progress per BOP VQ-6, CONTAINMENT MINI-PURGE OPERATION.
- 1VQ05C, Mini-flow Prg Exh fan, is running.

Subsequently a SI occurs on Unit 1.

Which of the following describes the 1VQ05C fan response to the SI, if any?

A. Will TRIP on Low ΔP

B. Will TRIP on VQ damper interlocks

C. Will NOT trip and will remain running.

D. Will NOT trip, but should be manually shut down

Answer	B

Answer Explanation

A – Plausible: Because a low fan DP will bring in the same alarm window as a fan trip. However, the low DP is a separate input to the alarm and not a fan trip interlock.

B – Correct: An SI signal will cause the mini-purge exhaust valves to close (on phase A) and the exhaust fan is interlocked to all three valves to trip if any of the valves is not full open.

C – Plausible: Because there is no direct trip signal between a phase a containment isolation and the mini-purge exhaust fan. Also plausible because the Mini-Purge exhaust fan suction is from the same inside containment damper as the Post-LOCA exhaust Fan (which may be used during an event where SI is required).

D – Plausible: Because there is no direct trip signal between a phase a containment isolation and the mini-purge exhaust fan.

Question Information

Topic	Byron 2023 NRC Exam Q57				
User ID	RS20029-N02			**System ID**	2498002
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	6E-1-4030VQ05 Rev. K BAR 0-32-C2 Rev. 52		
Training Objective	S.VP1-08-F		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requires examinee knowledge of the cause and effect relationship between an SI signal and the containment purge system.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.029.K1.03	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp:
Knowledge of the physical connections and/or cause and effect relationships between the Containment Purge System and the following systems: (CFR: 41.7 to 41.9 / 45.8) ESFAS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 58	ID: 2498176	Points: 1.00

- The Spent Fuel Pit system is in operation with the U-1 Spent Fuel Pit Pump (1FC01P) running.
- Subsequently the U-1 Spent Fuel Pit Pump Cubicle Cooler fans TRIP.

With NO operator action, what will the result of the above condition?

A. 1FC01P will trip immediately.

B. 1FC01P will trip when the room temperature reaches the alarm setpoint.

C. U-2 Spent Fuel Pit Pump Cubicle Cooler fans auto start immediately.

D. U-2 Spent Fuel Pit Pump Cubicle Cooler fans auto start when the room temperature reaches the alarm setpoint.

Answer	D

Answer Explanation

A – Plausible: Because a pump trip when a support equipment trip is a logical conclusion (on non-safety related equipment)

B – Plausible: Because high room temperature is an interlock for an auto start of a cubicle cooler and a high temperature alarm in the MCR.

C – Plausible: Because the U-2 SFP pump cubicle coolers start immediately when the U-2 SFP pump is started.

D – Correct: The SFP Pumps cubicles are separated by a barrier wall. However, the rooms are adjacent and do not have a door between them that separates the area air flow (they are not safety related equipment). When a SFP pump is running without the associated cubicle cooler, both rooms will heat up from the running pump motor. The cubicle coolers will auto start on high room temperature, even with the associated pump not running.

Question Information

Topic	Byron 2023 NRC Exam Q58				
User ID	RS20033-N06			System ID	2498176
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	Difficulty	N/A
Technical Reference and Revision #	6E-1-4030FC01 Rev. E 6E-1-4030VA16 Rev. M		
Training Objective	S.VA1-05-A		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requires examinee knowledge of the effect on the SFP system during a PVS malfunction.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.033.K6.10	Safety Function	Tier 2	Group	RO Imp: 2.5	SRO Imp:
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Spent Fuel Pool Cooling System: (CFR: 41.7 / 45.7) PVS					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

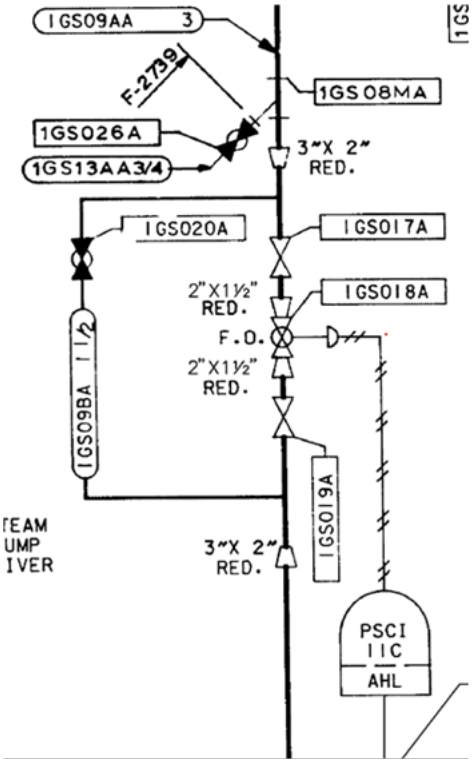
Cross Reference Links

None

Question 59	ID: 2501238	Points: 1.00

- Unit 1 is at 100% power.
- A tagout is being prepared to remove 1GS018A, LP Turbine GS seal PCV, from service for valve internals replacement.
- The isolation points are 1GS017A and 1GS019A.
- Special instructions direct 1GS020A throttled open to supply GS to the turbine gland seal.

(see P&ID below)



What will be the tagout EXCEPTIONAL status and why?

A. The tagout is NOT exceptional because no exceptional criteria is met.

B. The tagout IS exceptional because of GS system temperature AND pressure.

C. The tagout IS exceptional because of GS system temperature AND vent/drain availability.

D. The tagout IS exceptional because of GS system pressure AND vent/drain availability.

Answer	C

Answer Explanation

A – Plausible: Because the 1GS018A is completely isolated, but not dual isolation from a high temperature system.

B – Plausible: Because the gland steam system is supplied by main steam which is above the 500 psig criteria prior to being reduced to 118.5 psig by 1GS002, GS pressure regulator. Therefore, the upstream pressure at 1GS018A is normally 118.5 psig and below the 500 psig criteria.

C – Correct: Per OP-AA-109-101, the applicable criteria for an exceptional tagout is lack of dual valve isolation WITH a vent or drain path when isolating >500 psig or >200°F. Also, If valves are not available to drain/depressurize an isolated system, then DESIGNATE the tagout as exceptional. The existing drain valve (1GS026A is outside the isolation boundaries and the GS system is a saturated steam system (>200°F).

D – Plausible: Because the gland steam system is supplied by main steam which is above the 500 psig criteria prior to being reduced to 118.5 psig by 1GS002, GS pressure regulator. Therefore, the upstream pressure at 1GS018A is normally 118.5 psig and below the 500 psig criteria.

Question Information

Topic	Byron 2023 NRC Exam Q59				
User ID	RS20045-N05			**System ID**	2501238
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	OP-AA-109-101 Rev. 18 P&ID M-35 sht 5A Rev. AC		
Training Objective	T.AM33-11		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requires examinee knowledge of the tagout process on a Main Turbine auxiliary system.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

P2.2.13	Safety Function	Tier	Group	RO Imp: 4.1	SRO Imp: 4.3
Knowledge of tagging and clearance procedures (CFR: 41.10 / 43.1 / 45.13)					
G.SYS.045	Safety Function	Tier 2	Group	RO Imp:	SRO Imp:
MT/G Main Turbine Generator System					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 60	ID: 2498220	Points: 1.00

- Both units are at 100% Power.
- 0A Control Room Chiller is running.
- 0B Control Room Chiller is in standby.

The following then occurs:
- A loss of off site power on Unit 1 ONLY.
- Bus 141 faults and de-energizes.

Two minutes later (assuming no operator action) the 0A Control Room Chiller is ____(1)____ and the 0B Control Room Chiller is ____(2)____.

A. (1) running
(2) running

B. (1) running
(2) not running

C. (1) not running
(2) running

D. (1) not running
(2) not running

Answer	C

Answer Explanation

A – Plausible: Because both VC chillers would be running if bus 141 had not faulted.

B – Plausible: Because the 0A VC chiller would be running if bus 141 had not faulted. Also a common misconception is the 0B VC chiller is powered from Unit 2. The other unit common 4kV bus loads (WS pps, VA fans, VA chillers, service building chillers) are all divided between Unit 1 and Unit 2 busses. Unit 2 did not experience a LOOP (no DG sequencing of loads), therefore with the misconception that U-2 powers the 0B VC chiller, the 0B VC chiller would not have started.

C – Correct: 0A VC chiller would not be running because bus 141 faulted and locked out loads. 0B VC chiller is running because bus 142 would have been re-energized by the 1B DG and sequenced on the ESF loads, including the 0B VC chiller.

D – Plausible: Because a common misconception is the 0B VC chiller is powered from Unit 2. The other unit common 4kV bus loads (WS pps, VA fans, VA chillers, service building chillers) are all divided between Unit 1 and Unit 2 busses. Unit 2 did not experience a LOOP (no DG sequencing of loads), therefore with the misconception that U-2 powers the 0B VC chiller, the 0B VC chiller would not have started.

Question Information

Topic	Byron 2023 NRC Exam Q60				
User ID	RS20050-N02			**System ID**	2498220
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

NRC Exams Only			
Question Type	Modified from Bwd ILT systems exam bank.	**Difficulty**	N/A
Technical Reference and Revision #	6E-0-4030WO01 Rev. P 6E-0-4030WO02 Rev. P		
Training Objective	S.W01-04-A		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requires examinee knowledge of the VC chiller units power supplies.
SRO-Only Justification	Not applicable
Additional Information	None

Original question:
Given the following:

- Unit One is at 100% Power, all systems in automatic
- 0A Control Room Chiller is running
- 0B Control Room Chiller is in standby
- SAT 142-1 develops a fault
- All systems respond normally
- Both Diesel Generators subsequently reenergize both ESF buses

Two minutes later (assuming no operator action) the 0A Control Room Chiller is _____ and the 0B Control Room Chiller is _____.

A. running running

B. running not running

C. not running running

D. not running not running

Answer	**A**

K/A Reference(s)

SYS.050.K2.02	Safety Function	Tier 2	Group	RO Imp: 3.1	SRO Imp:
Knowledge of electrical power supplies to the following: (CFR: 41.7) Chiller units					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 61	ID: 2498224	Points: 1.00

- Unit 1 is at 100% power.
- A recent RCS leakrate surveillance indicated 0.015 GPM rise in RCS leakage.
- The US has directed the RO to monitor RMS indications for a possible SG tube leak.

The FIRST rad monitor alarm indicating a small SG tube leak will be from the...

A. 0PR16J, B/D AFTER FILTER 0A OUTLET.

B. 1PR27J, SJAE GS EXH.

C. 1PR028J, AUX BLDG VENT STACK EFFLUENT MONITOR.

D. 1AR022/023, MS LINE.

Answer	B

Answer Explanation

A – Plausible: Because the SG blowdown after filter RAD monitor 0PR16J will alarm on SG tube leakage and redirect flow to the BD monitor tank making this a plausible distractor. The blowdown after filter RAD alarm is delayed in seeing the leakage because it must cycle through the steam system back through feed and mix with the SG water then pass through the demins before it is sensed by the after filter alarm. Additionally, the 0PR16J monitors water downstream of the blowdown demins which will remove much of the radioactivity from the secondary water prior to the 0PR16J monitoring. Therefore, this will not be the first alarm on the radiation monitoring system.

B – Correct: The air ejector RAD monitor is designed to detect small leakage amounts in the SG tubes. This is the first alarm indication of a SG tube leak. RE-PR027 monitors condenser air removal at exhaust of SJAEs. This sample location allows us to detect as small as ~ 5 gal/day primary-to-secondary leakage from the SJAEs.

C – Plausible: Because the 1PR028J, Aux Bldg Vent Stack will also monitor the off gas flow through the Unit 1 plant vent stack. However, the 1PR028J are designed to detect high activity from waste gas or containment releases, not small SG tube leaks.

D – Plausible: Because the main steam line rad monitors will sense minor SG tube leaks. However, these are Safety Related Tech Spec monitors that are used to identify steam generator tube ruptures and typically will not alarm until larger tube leaks exist.

Question Information

Topic	Byron 2023 NRC Exam Q61				
User ID	RS20055-N01			**System ID**	2498224
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

NRC Exams Only			
Question Type	Bank from Bwd 2018 NRC exam.	**Difficulty**	N/A
Technical Reference and Revision #	Byron ILT LP I1-OG-XL-01, Condenser Air Removal Rev. 7 1BOA SEC-8 Rev. 112		
Training Objective	S.AR1-06		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A by requires examinee knowledge of the design features of the CARS effluent monitoring.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.055.K4.02	Safety Function	Tier 2	Group	RO Imp: 3.3	SRO Imp:
Knowledge of Condenser Air Removal System design features and/or interlocks that provide for the following: (CFR: 41.7) Effluent control and monitoring					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 62	ID: 2498253	Points: 1.00

- Unit 1 is in Mode 5.
- The CD/CB system is in recirculation operation through the high pressure clean up loop.
- The Condensate Polishers are online at maximum flow directed by chemistry.

The following then occurs:
- Local annunciators at 0CP01J, MB POLISHER 0A and 0B DIFF PRESS HIGH, alarm.
- MCR annunciator 1-17-D9, CB PUMP SUCTION PRESS LOW, alarms.
- CB pump suction pressure is 34 psig and slowly DROPPING.

With the above conditions, the...

A. Standby CD/CB pump AUTO STARTS.

B. 1CD152A/B, CD PUMPS RECIRC VLV CLOSES.

C. 1CD157A/B, GS CNDSR BYPASS VALVES OPEN.

D. 1CD210A/B, CNDS POLISHING BYPASS VLVS OPEN.

Answer	D

Answer Explanation

A – Plausible: Because the standby CD/CB pump will get an auto start signal from FW pump low NPSH, but not low CB pump suction pressure.

B – Plausible: Because 1CD152A/B get an auto close signal from FW pump low NPSH, but not low CB pump suction pressure.

C – Plausible: Because 1CD157A/B get an auto open signal from FW pump low NPSH, but not low CB pump suction pressure.

D – Correct: With the stem conditions, the condensate polishers are clogging and restricting flow to the CB pumps suction. MCR alarm 1-17-D9 setpoint is CB pump suction pressure at 40 psig. Low CB pump suction pressure of 35 psig will cause 1CD210A/B valves to open and bypass the condensate polishers.

Question Information

Topic	Byron 2023 NRC Exam Q62				
User ID	RS20056-N02			**System ID**	2498253
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

NRC Exams Only			
Question Type	New for Byron NRC exam.	**Difficulty**	N/A
Technical Reference and Revision #	BAR 1-17-D9 Rev. 2		
Training Objective	S.CD1-07-D		

Previous NRC Exam Use	None

References Provided	None
K/A Justification	This question meets the K/A requiring examinee ability to monitor changes associated with CD pressure during condensate polisher operations.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.056.A1.09	Safety Function	Tier 2	Group	RO Imp: 2.5	SRO Imp:
Ability to predict and/or monitor changes in parameters associated with operation of the Condensate System, including: (CFR: 41.5 / 45.5) Long-cycle recirculation parameters (temperature, pressure, and flow level)					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 63 ID: 2501711 Points: 1.00

- A fire is occurring in the Upper Cable Spreading Room directly above the Unit 1 MCR.
- All detectors in the area are in alarm on 1PM09J, Fire Detection panel.

(1) With the above conditions, the fire suppression system will AUTOMATICALLY actuate

(2) Per BAP 1100-10, RESPONSE PROCEDURE FOR FIRE / FIRE ALARM, the MCR operator will IMMEDIATELY call...

A. (1) CO_2 suppression.
(2) Security.

B. (1) CO_2 suppression.
(2) Plant Engineering.

C. (1) Halon suppression.
(2) Security.

D. (1) Halon suppression.
(2) Plant Engineering.

Answer	**C**

Answer Explanation

A - Plausible: Because CO2 is a backup system in the UCSR that requires manual actuation.

B - Plausible: Because CO2 is a backup system in the UCSR that requires manual actuation. Also procedure direction to call Plant Engineering is routinely used in many plant abnormal or emergency situations including LOCAR entries, BOAs and BEPs.

C - Correct: Halon will auto actuate on 2/2 coincidence of POC and thermal detectors in the upper cable spreading room. If CO2 or Halon has actuated in the UCSR, BAP 1100-10, step C.1.b.5).a) directs the control room operator to immediately call Security to ask for a computer access search for personnel in the affected areas.

D - Plausible: Because procedure direction to call Plant Engineering is routinely used in many plant abnormal or emergency situations including LOCAR entries, BOAs and BEPs.

Question Information

Topic	Byron 2023 NRC Exam Q63				
User ID	RE20067-N01			**System ID**	2501711
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC Exams Only			
Question Type	Bank from 2011 Bwd NRC exam.	**Difficulty**	N/A
Technical Reference and Revision #	ILT lesson plan N-BY-TQ-ILT-FP-57 Rev. 13 BAR 0-37-A4 Rev. 12 BAP 1100-10 Rev. 21		
Training Objective	3C.FP-01-B		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A requiring examinee ability to predict impact of fire in the plant on the FP system and use procedure to mitigate the consequences.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

SYS.086.A2.05	Safety Function	Tier 2	Group	RO Imp: 4.0	SRO Imp: 3.8
Ability to (a) predict the impacts of the following on the Fire Protection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) Fire in the plant					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 64 ID: 2498327 Points: 1.00

An RO took the shift as the Unit RO while working a 12-hour schedule as indicated below:

Sep. 07, 2023 - OFF
Sep. 08, 2023 - 0700 to 1900.
Sep. 09, 2023 - 0700 to 2100.
Sep. 10, 2023 - 0700 to 1900.
Sep. 11, 2023 - OFF
Sep. 12, 2023 - 0700 to 2300.
Sep. 13, 2023 - 0900 to 1900.
Sep. 14, 2023 - 0700 to 1900.

The RO FIRST violated Work Hour Limits on …
(Assume the average hours from previous weeks are not violated by the above hours.)

A. Sep. 9, 2023

B. Sep. 10, 2023

C. Sep. 12, 2023

D. Sep. 14, 2023

Answer **D**

Answer Explanation

A – Plausible: Because the shift was extended to 14 hours. However, allowable work period is 16 hours in 24.

B – Plausible: Because the break period was reduced to 10 hours. However, 10 hours is the minimum required break period.

C – Plausible: Because the shift was extended to 16 hours. However, allowable work period is 16 hours in 24.

D – Correct: The RO worked more than 72 hours in a 7 day period violating 10 CFR 26 Work Hour Limits.
Per LS-AA-119, section 5.1.1. The following limits apply to covered individuals regardless of unit status:

No more than 16 work hours in any 24-hour period
No more than 26 work hours in any 48-hour period
No more than 72 work hours in any 7-day (168-hour) period
At least a 10-hour break between successive work periods, or an 8-hour break when a break of less than 10 hours is necessary to accommodate a crew's scheduled transition between work schedules or shifts
A 34-hour break in any 9-day period (this limit may be incorporated into minimum days off requirements)

Question Information

Topic	Byron 2023 NRC Exam Q64				
User ID	RG10215-C14-01			**System ID**	2498327
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from Bwd 2016 Cert exam.	**Difficulty**	N/A
Technical Reference and Revision #	LS-AA-119 Rev. 16		
Training Objective	WHR-OVR-04		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A requiring examinee to use procedures related to shift staffing with relation to work hour rules. Per 10 CFR 26.205 Individuals are responsible for: Managing their work hours to prevent impairment from fatigue and violation of the rule.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

P2.1.05	Safety Function	Tier 3	Group	RO Imp: 2.9	SRO Imp: 3.9
Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations (reference potential) (CFR: 41.10 / 43.5 / 45.12)					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 65	ID: 2498329	Points: 1.00

The RUNNING SX pump TRIPS.

Per OP-BY-101-0004, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, the RO will…

A. take the Prompt Action and THEN announce the SX pump trip. NO US repeat back is required.

B. announce the SX pump trip and THEN take Prompt Action. NO US repeat back is required.

C. take the Prompt Action and THEN announce the SX pump trip, the US WILL repeat back the communication.

D. announce the SX pump trip and the intent to perform prompt response actions, the US WILL repeat back the communication, and THEN the NSO will take the Prompt Action.

Answer	D

Answer Explanation

A – Plausible: Because communication standards are sometimes lowered during transient situations. It's incorrect because communication will take place before taking prompt action, and US repeat back is still required.

B – Plausible: Because communication standards are sometimes lowered during transient situations. It's incorrect because repeat back from US is still required.

C – Plausible: Because communication standards are sometimes lowered during transient situations. It's incorrect because communication will take place before taking prompt action.

D – Correct: OP-BY-101-0004, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION, step 4.2, Prompt Action Communication states when a transient requiring PROMPT OPERATOR ACTION occurs the response actions are governed as follows: The Reactor Operator (RO) will announce the transient requiring prompt operator action, followed by a proper repeat back by the Unit Supervisor validating the action to be taken is correct (preferred method), THEN take the prompt action.

Question Information

Topic	Byron 2023 NRC Exam Q65				
User ID	RG10217-N01			System ID	2498329
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from Bwd 2020 NRC exam.	Difficulty	N/A

Technical Reference and Revision #	OP-BY-101-0004, Rev. 30
Training Objective	T.AM69-01-B
Previous NRC Exam Use	None

References Provided	None
K/A Justification	This question meets the K/A requiring the examinee must have the ability to make proper communications during transient situations as described in OP-BY-101-0004, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

P2.1.17	Safety Function	Tier 3	Group	RO Imp: 3.9	SRO Imp: 4.0
Ability to make accurate, clear, and concise verbal reports (CFR: 41.10 / 45.12 / 45.13)					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 66 ID: 2498335 Points: 1.00

Why is the LO-2 SG level reactor trip setpoint different between Unit 1 and Unit 2?

A. Difference in SG design and level tap locations.

B. Difference in the rated thermal power of each unit.

C. Difference in the Aux feed water systems ability to deliver water to the SGs.

D. Difference in the Main feed water systems ability to deliver water to the SGs.

Answer	**A**

Answer Explanation

A – Correct: The LO-2 SG level reactor trip setpoint difference between units (Unit 1 at 18% vs. Unit 2 at 36.3%) is due to the difference in design (B&W vs D-5) and the relative locations of the narrow range level taps. Both of these setpoints represent approximately the same mass of secondary water in the S/G.

B – Plausible: Because a common misconception is that the units have a difference in rated thermal power. However, per tech specs the rated thermal power is the same (3645 MWt) for each unit. The misconception is based upon the different 100% power electrical MW output which is primarily due to the difference in the efficiency of the unit's different steam generators.

C – Plausible: Because the AF water system flow is directed through the FW tempering line on both units. However, Unit 1 tempering flow enters the SGs through the Main FW nozzle, while Unit 2 tempering flow enters the SG through the upper tempering line nozzle.

D – Plausible: Because the Main feed water system flow is directed into the Unit 1 SGs through the Main FW nozzle at the upper level of the SG and then through a goose necked ring header above the SG tubes, while Unit 2 Main feed water nozzle enters the SGs at the lower level into a preheater section just above the tube sheet.

Question Information

Topic	Byron 2023 NRC Exam Q66				
User ID	RG20004-N02			**System ID**	2498335
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Byron ILT SG lesson plan N-BY-TQ-ILT-SG-22 Rev. 9		
Training Objective	S.SG-04-A		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A requiring the examinee ability to explain variations in unit systems.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

P2.2.04	Safety Function	Tier 3	Group	RO Imp: 3.6	SRO Imp: 3.6
(Multi-unit license) Ability to explain the variations in control room layouts, systems, instrumentation, and/or procedural actions between units at a facility (CFR: 41.6 / 41.7 / 41.10 / 45.1 / 45.13)					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 67	ID: 2498339	Points: 1.00

When plant conditions require entry into Tech Spec LCO 3.0.3, what is the maximum time allowed to PLACE the unit in Mode 3?

A. 6 Hours

B. 7 Hours

C. 12 Hours

D. 13 Hours

Answer	B

Answer Explanation

A – Plausible: Because when a required action and associated completion time is not met in multiple Tech Spec LCOs, the unit must be in Mode 3 within 6 hours.

B – Correct: per LCO 3.0.3 action must be initiated within 1 hour to place the unit in Mode 3 within 7 hours.

C – Plausible: Because per LCO 3.0.3 action must be initiated within 1 hour to place the unit in Mode 4 within 13 hours. Discounting the 1 hour of preparation to ramp, the unit has 12 hours (ramp down time) to be in Mode 4. Additionally, several other Tech Spec action times are 12 hours.

D – Plausible: Because per LCO 3.0.3 the unit must be in Mode 4 within 13 hours.

Question Information

Topic	Byron 2023 NRC Exam Q67				
User ID	RG20022-N02			System ID	2498339
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	Difficulty	N/A
Technical Reference and Revision #	Tech Spec 3.0 Amendment 216		
Training Objective	S.TS1-05-D		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A requiring knowledge of LCO 3.0.3.
SRO-Only Justification	Not applicable
Additional Information	The question is RO level because action must be initiated within 1 hour for Tech Spec compliance.

K/A Reference(s)

P2.2.22	Safety Function	Tier 3	Group	RO Imp: 4.0	SRO Imp: 4.7
Knowledge of limiting conditions for operation and safety limits (CFR: 41.5 / 43.2 / 45.2)					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 68 ID: 2498357 Points: 1.00

A safety-related area radiation monitor can be operated from __(1)__ and can be monitored from __(2)__.

A. (1) ONLY its RM-23
(2) its RM-23 and RMS

B. (1) ONLY its RM-23
(2) ONLY its RM-23

C. (1) its RM-23 and RMS
(2) its RM-23 and RMS

D. (1) its RM-23 and RMS
(2) ONLY its RM-23

Answer	A

Answer Explanation

A – Correct: ONLY its RM-23, its RM-23 and RMS is correct. Safety related monitors can only be operated at the associated RM-23. They can be monitored at both the RM-23 and RMS.

B – Plausible: Because this is a common misconception among novice applicants due to the difference between these controls in the simulator and the plant. These rad monitors can also be monitored at RMS.

C – Plausible: Because this is a common misconception among novice applicants due to the difference between these controls in the simulator and the plant. Safety related monitors cannot be operated from RMS.

D – Plausible: Because this is a common misconception among novice applicants due to the difference between these controls in the simulator and the plant. Safety related monitors cannot be operated from RMS.

Question Information

Topic	Byron 2023 NRC Exam Q68				
User ID	RG-2.3.15-72			**System ID**	2498357
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

NRC Exams Only			
Question Type	Bank from 2019 Bwd NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	BOP AR/PR-6 Rev. 2		
Training Objective	S.AR1-03-B		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A requiring knowledge of radiation monitoring systems and where they can be operated.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

P2.3.05	Safety Function	Tier 3	Group	RO Imp: 2.9	SRO Imp: 2.9
Ability to use RMSs, such as fixed radiation monitors and alarms or personnel monitoring equipment (CFR: 41.11 / 41.12 / 43.4 / 45.9)					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 69	ID: 2498383	Points: 1.00

- A MCR evacuation is required.
- An RO is dispatched to activate the Remote Shutdown Panel (RSP).

PRIOR to manipulating the Remote/Local switch the RO will check _____(1)_____.

To activate the RSP, the RO will place the equipment Remote/Local switches to ___(2)___.

A. (1) the component is shut down or closed.
(2) local

B. (1) the RSP control switch matches current component position.
(2) local

C. (1) the component is shut down or closed.
(2) remote

D. (1) the RSP control switch matches current component position.
(2) remote

Answer	B

Answer Explanation

A – Plausible: Because control switches for motor driven equipment in the RSP are generally kept in the stop position unless they have a spring return to center position (i.e. equipment with auto start functions).

B – Correct: Transfer of equipment control to the RSP requires the remote/local switches to be in local. A note in 1/2BOA PRI-5 reads "Check control switch position matches current component position prior to placing Local/Remote switch in the Local position."

C – Plausible: Because the remote/local switches are located in the "Remote" Shutdown Panel. A common misconception is the switches need to be in remote for control from the Remote Shutdown Panel. Additionally control switches for motor driven equipment in the RSP are generally kept in the stop position (unless they have a spring return to center position).

D – Plausible: Because the remote/local switches are located in the "Remote" Shutdown Panel. A common misconception is the switches need to be in remote for control from the Remote Shutdown Panel.

Question Information

Topic	Byron 2023 NRC Exam Q69				
User ID	RG40034-N02			System ID	2498383
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only

Question Type	New for 2023 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA PRI-5 Rev. 111		
Training Objective	T.OA16-04		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A requiring knowledge of RO responsibilities outside the MCR during an emergency.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

P2.4.34	Safety Function	Tier 3	Group	RO Imp: 4.2	SRO Imp: N/A
Knowledge of RO responsibilities outside the main control room during an emergency (CFR: 41.10 / 43.5 / 45.13)					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 70 ID: 2498390 Points: 1.00

- Unit 1 was at 100% power and equilibrium Xenon.
- A step power reduction from 100% to 90% occurs due to an equipment malfunction.

With the above conditions, the INITIAL change in Xenon concentration will be PRIMARLY a:

A. reduction in the loss of Xenon due to neutron absorption.

B. rise in Xenon removal due to decay to Cesium.

C. rise in Xenon production from Iodine-135 decay.

D. reduction in Xenon produced directly by fission.

Answer	**A**

Answer Explanation

A – Correct: At equilibrium conditions Xenon production directly from fission (minor factor) and Iodine decay (major factor) is balanced with reduction by Xenon decay to cesium and burnout by neutron absorption. With a step reduction in Reactor power,

Xenon produced directly from fission will lower. However, production from 100% power Iodine level decay which is the major source of Xenon at power will continue to occur (only slowing). The burnout of Xenon will lower directly with power/flux and Xenon decay will remain unchanged. The overall effect is that due to continued delayed production of Xenon from Iodine decay and lower burnout Xenon levels will rise initially.

B – Plausible: Because Xenon decay is one means of Xenon removal. Xenon concentration will rise on a down power due the reduction of burnout with continued production by Iodine Decay. Xenon removal by decay to Cesium will remain relatively constant.

C – Plausible: Because this is the major factor in Xenon production. With a down power, Iodine production from fission will LOWER, Xenon will however continue to be produced at the same rate immediately following the power reduction due to Iodine decay from previous fissions.

D – Plausible: Because Xenon produced by fission does lower on a down power. Xenon concentration will rise on a down power due to the reduction of burnout and the continued production by Iodine decay.

Question Information

Topic	Byron 2023 NRC Exam Q70				
User ID	RT20006-N02			**System ID**	2498390
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

NRC Exams Only			
Question Type	Bank from Constellation Fundamentals Question bank	**Difficulty**	N/A
Technical Reference and Revision #	192006 INPO Operator Generic Fundamentals Reactor Theory – Fission Product Poisons Rev. 3.1		
Training Objective	192006 ELO 1.3		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	This question meets the K/A requiring knowledge of the removal of xenon following a down power event.
SRO-Only Justification	Not applicable
Additional Information	None

K/A Reference(s)

192006.K1.04	Safety Function	Tier 4	Group	RO Imp: 2.8	SRO Imp:
Describe the removal of xenon-135					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 71	ID: 2500361	Points: 1.00

Assume typical Byron breaker to breaker unit operations following a refueling outage.

Boron concentration will __(1)__ from the middle of core life to the end of core life PRIMARILY due to _____(2)_____.

A. (1) rise
(2) burnable poison burnout

B. (1) drop
(2) burnable poison burnout

C. (1) drop
(2) fuel depletion

D. (1) rise
(2) fuel depletion

Answer	C

Answer Explanation

A - Correct: The depletion of fuel is a negative reactivity component. Therefore, positive reactivity must be added by decreasing boron concentration to maintain a net zero reactivity in the core.

B - Plausible: Because burnable poison burnout is a factor in boron concentration requirements. After the initial drop in boron concentration (due to the build up of short and long lived fission products) burnable poison burnout is primarily responsible for the rise in boron concentration requirements.

C - Plausible: Because boron concentration rises during the fuel cycle. However, that occurs during the BOL to MOL period.

D - Plausible: Because boron concentration rises during the fuel cycle. However, that occurs during the BOL to MOL period. Also burnable poison burnout is a factor in boron concentration requirements. After the initial drop in boron concentration (due to the build up of short and long lived fission products) burnable poison burnout is primarily responsible for the rise in boron concentration requirements.

Question Information

Topic	Byron 2023 NRC Exam Q71				
User ID	RT2007-N02			**System ID**	2500361
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	192007 INPO Operator Generic Fundamentals Reactor Theory - Fuel Depletion and Burnable Poisons, Rev. 3.2 BCB Figure 11 Rev. 35		
Training Objective	ELO 1.3		
Previous 2 Byron NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of how and why boron concentration changes over core life.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

192007.K1.04	Safety Function	Tier 4	Group	RO Imp: 3.4	SRO Imp:
Describe how and why boron concentration changes over core life					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 72	ID: 2498394	Points: 1.00

- A Unit 1 reactor start up is in progress per 1BGP 100-2A1, REACTOR STARTUP after an inadvertent unit trip.
- The RO is monitoring SR counts for the eight-fold count rate hold point.

What were the plant conditions when the INITIAL source range counts were logged for EIGHT-FOLD CALCULATION on 1BGP 100-2T3, REACTOR STARTUP FLOW CHART?

A. All Shutdown banks fully INSERTED and RCS boron concentration at the 200°F Shutdown Margin Requirement.

B. All Shutdown banks fully INSERTED and RCS boron concentration at the ECC Critical Boron Concentration.

C. All Shutdown banks fully WITHDRAWN and RCS boron concentration at the 200°F Shutdown Margin Requirement.

D. All Shutdown banks fully WITHDRAWN and RCS boron concentration at the ECC Critical Boron Concentration.

Answer	D

Answer Explanation

A - Plausible: Because source range counts are recorded during these conditions. However, they are used to determine SR count rises during dilution to critical boron concentration (no more than a factor of 3) and during shutdown bank withdrawal (no more than a factor of 2.5).

B - Plausible: Because source range counts are recorded during this condition. However, they are used to determine SR count rises during shutdown bank withdrawal (no more than a factor of 2.5).

C - Plausible: Because source range counts are recorded during these conditions. However, they are used to determine SR count rises during dilution to critical boron concentration (no more than a factor of 3).

D - Correct: Initial RCS boron concentration for eight-fold calculation is logged per 1BGP100-2A1, step F.4. This is after boron concentration is adjusted to ECC predicted critical boron concentration (1BGP 100-2, step F.7 or F.14) and after shut down bank withdrawal (1BGP 100-2, step F.13). 1BGP 100-2A1 is then performed per 1BGP 100-2, step F.15.

Question Information

Topic	Byron 2023 NRC Exam Q72				
User ID	RT20007-N02			**System ID**	2498394
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BGP 100-2A1 Rev. 38		

	1BGP 100-2 Rev. 54
Training Objective	T.GP02-02
Previous 2 Byron NRC Exams Use	None

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of parameters to monitor during reactor start up.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

192008.K1.01	Safety Function	Tier 4	Group	RO Imp: 3.5	SRO Imp:
List parameters that should be monitored and controlled during the approach to criticality					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 73 ID: 2498403 Points: 1.00

- A LOCA has occurred on Unit 1 following normal 100% power operation.
- Wide Range HOT LEG MCB Indicators are reading 550°F.

What is the LOWEST RCS pressure that will still maintain 20°F of subcooling?

A. 885 PSIG

B. 900 PSIG

C. 1211 PSIG

D. 1241 PSIG

Answer	**C**

Answer Explanation

A - Plausible: Because 885 is the saturation pressure in PSIA for 530°F (550-20).

B - Plausible: Because 900 PSIG is the saturation pressure for 530°F (550-20).

C - Correct: To obtain 20°F RCS subcooling, the operator must add 20°F to the current RCS temperature of 550°F to obtain 570°F. The saturation pressure for 570°F is 1226 PSIA. Convert 1226 PSIA to PSIG by subtracting 15 psi equals 1211 PSIG Maintaining the RCS above 1211 PSIG will ensure the RCS is at least 20°F subcooled for the current temperature of 550°F.

D - Plausible: Because 1241 is 1226+15 PSI (did not convert to from PSIA to PSIG correctly).

Question Information

Topic	Byron 2023 NRC Exam Q73				
User ID	RT30003-N01			**System ID**	2498403
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

NRC Exams Only			
Question Type	Modified from Byron 2022 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	193003 INPO Operator Generic Fundamentals Thermodynamics – Steam Rev. 4.1, Steam Tables		
Training Objective	193003 ELO 2.4		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of the definition of subcooled.
SRO-Only Justification	N/A
Additional Information	None

· A LOCA has occurred following normal 100% power operation.
· Wide Range HOT LEG MCB Indicators are reading 590°F.

What is the LOWEST RCS pressure that will still maintain 20°F of subcooling?

A. 1212 PSIG

B. 1226 PSIG

C. 1631 PSIG

D. 1646 PSIG

Answer: D

K/A Reference(s)

193003.K1.16	Safety Function	Tier 4	Group	RO Imp: 2.7	SRO Imp:
Define the following terms: subcooled and compressed liquids					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 74	ID: 2498443	Points: 1.00

- Unit 1 operators have indications that a PZR safety valve is partially open.
- PZR pressure is 1985 PSIG.
- PRT pressure at 10 PSIG.

The expected tail pipe temperature for the open PZR safety valve would be ____°F.

A. 270

B. 240

C. 195

D. 175

Answer	B

Answer Explanation

A - Plausible: Because this is where the 1136 enthalpy line intersects the 10 psia constant pressure line and then moves up to the saturation curve along the constant entropy line (approx 1.68 entropy). The error is following a constant entropy line up to the saturation line instead of following the 25 psia constant pressure line up to the saturation line.

B - Correct: With PZR at 1985 PSIG (2000 PSIA), the enthalpy of the steam is 1136 btu/lbm. 100% quality steam undergoing an isenthalpic transition to 10 PSIG (25 PSIA) will result in steam at 97.5% quality at 240°F. The throttling process is isenthalpic. Using the Mollier diagram, first locate the point where the 2000 psia constant pressure curve intersects the saturation line. Then move to the right along the 1136 enthalpy line until intersecting the 25 psia constant pressure line. Then follow the 25 psia constant pressure line to intersect the saturation curve. The 240°F constant temperature line intersects at the same point.

C - Plausible: Because this is where the 1136 enthalpy line intersects the 10 psia constant pressure line and then moves up to the saturation curve. The error is not converting the 10 psig to 25 psia.

D - Plausible: Because this is where the 1136 enthalpy line intersects the saturation line. The error is not following the 25 psia constant pressure line up to the saturation line.

Question Information

Topic	Byron 2023 NRC Exam Q74				
User ID	193004 K1.15 P76			**System ID**	2498443
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

NRC Exams Only			
Question Type	Bank from Constellation Fundamentals exam bank	**Difficulty**	N/A
Technical Reference and Revision #	193004 INPO Operator Generic Fundamentals Thermodynamics – Steam Rev. 4.1, Thermodynamic Processes Rev. 3.1		
Training Objective	193004 ELO 1.5		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring candidate to determine the exit conditions for a throttling process for a steam system.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

193004.K1.15	Safety Function	Tier 4	Group	RO Imp: 2.8	SRO Imp:
Determine the exit conditions for a throttling process based on the use of steam and/or water					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 75	ID: 2498453	Points: 1.00

- Unit 1 is at 50% power MOL.
- Axial Flux Difference (Delta I) is +0.5 above target.

U1 hot channel (peaking) factors will LOWER if _____ is RAISED.

A. Tave

B. core age

C. reactor power

D. axial flux difference (Delta I)

Answer	B

Answer Explanation

A - Plausible: Because lowering Tave would lower fuel temperatures and would thus lower peaking factors. Raising Tave raises RCS temperature and thus fuel temperatures for the same power.

B - Correct: RCS flow rate, RCS temperature, RCS Pressure, Rx Power/power density, and Fuel Heat transfer coefficient all affect peaking and hot channel factors. Clad creep and fuel densification affect the fuel heat transfer coefficient. Rod position/DI affect power density. At EOL clad creep and fuel pellet swell lowers fuel temperatures and therefore lowers hot channel factors.

C - Plausible: Because lowering reactor power would lower peaking factors. Raising reactor power raises hot channel factors by raising power density.

D - Plausible: Because actions that would shift DI closer to target would lower peaking factors. Lowering DI with it on target raises the power density in the core in an where it is not desired.

Question Information

Topic	Byron 2023 NRC Exam Q75				
User ID	RT30009-N01			System ID	2498453
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	RO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

NRC Exams Only			
Question Type	Bank from Constellation Fundamentals exam bank	Difficulty	N/A
Technical Reference and Revision #	193009 INPO Operator Generic Fundamentals		

	Thermodynamics – Core Thermal Limits Rev. 3.1
Training Objective	1930094 ELO 1.3
Previous NRC Exams Use	None

References Provided	None
K/A Justification	Meets the K/A by requiring candidate to determine factors affecting channel peaking factors.
SRO-Only Justification	N/A
Additional Information	None

K/A Reference(s)

193009.K1.07	Safety Function 4	Tier	Group	RO Imp: 3.3	SRO Imp:
Describe factors that affect peaking and hot channel factors					

Learning Objective(s)

2023 NRC RO Exam
User (Sys) ID N/A (1662931)

Cross Reference Links

None

Question 76	ID: 2498512	Points: 1.00

- Unit 1 was at 100% power prior to a reactor trip.
- 1BEP-0 step 4, CHECK SI STATUS, is in progress.
    - 1B SG NR level lowered rapidly to 0%.
    - 1B SG pressure is 700 psig and lowering slowly.
    - 1A/1C/1D SG NR levels are 21% and rising.
    - 1A/1C/1D SG pressures are 900 psig and stable.
    - AF is operating as required.
    - Containment pressure is 9 psig.
    - All Tcold are 537˚F and slowly lowering.
    - RCS pressure is 2000 psig and slowly lowering.

1. What malfunction has occurred?

2. What is the procedure flow path after 1BEP-0?

A. (1) Steam line break on 1B SG.
(2) 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT then 1BEP-2, FAULTED SG ISOLATION.

B. (1) Feed line break on 1B SG.
(2) 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT then 1BEP-2, FAULTED SG ISOLATION.

C. (1) Steam line break on 1B SG.
(2) 1BEP-2, FAULTED SG ISOLATION then 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

D. (1) Feed line break on 1B SG.
(2) 1BEP-2, FAULTED SG ISOLATION then 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

Answer	D

Answer Explanation

A - Plausible: Because 1BEP-1 has action to diagnose a secondary break and return to 1BEP-2. However, 1BEP-1 will not be the procedure to mitigate the transient, based on the lowering RCS temp and SG level 1BEP-0 will direct using 1BEP-2.

B - Plausible: Because a feed line fault is indicated by these conditions and 1BEP-1 does have steps to diagnose a secondary break and return to 1BEP-2. However, 1BEP-1 will not be the procedure to mitigate the transient, based on the lowering RCS temp and SG level 1BEP-0 will direct using 1BEP-2.

C - Plausible: Because 1BEP-2 will be the mitigating procedure and the steam line break will cause RCS temperature to lower. However, a steam line break will cause the affected SG pressure and level to lower more rapidly than the others.

D - Correct: A feedline break inside containment will cause RCS temperature to lower due to the SG water flashing to steam cooling the RCS. This effect also causes the pressure in that SG to stay elevated until the feed ring is uncovered. Once the steam generator is isolated and blown down the next procedure transition will be to 1BEP-1 to ensure no other issue exists before transitioning to SI termination.

Question Information

Topic	Byron 2023 NRC Exam Q76				
User ID	RE20059-N02			**System ID**	2498512
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only			
Question Type	Bank from Bwd 2018 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Byron ILT LP, N-BY-TQ-ILT-BEP-2, Rev.10 1BEP-0, Rev. 306		
Training Objective	T.EP03-02		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring the examinee to interpret the SG pressure drop and compare it to the indications provided to determine the event in progress and determine the appropriate procedures to mitigate the consequences.
SRO-Only Justification	The question is SRO only because it requires selection of emergency procedures and the flow path of the emergency procedures to mitigate the consequences of the event.
Additional Information	None

K/A Reference(s)

EPE.007.EA2.09	Safety Function	Tier 1	Group	RO Imp: 3.5	SRO Imp: 3.7
Ability to determine and/or interpret the following as they apply to a Reactor Trip: (CFR: 41.7 / 45.5 / 45.6) S/G pressure					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 77	ID: 2498520	Points: 1.00

- An RCS LOCA occurred on Unit 1 from full power.
- Bus 142 is damaged and CANNOT be energized.

1BEP ES-1.3, TRANSFER TO COLD LEG RECIRCULATION UNIT 1, step 7, RESET SI IF NECESSARY, is in progress,

- The 1A RH pump has tripped. 1A SI and CV pumps are running.
- RWST level is 8%.
- RVLIS plenum level is 55%.
- CETCs are 575 °F and slowly rising.
- Containment Floor Water Level is 24 inches.
- Containment Pressure is 21 psig with 1A CS train running.

What action will the crew perform NEXT?

A. Stop 1A ECCS and CS pumps AND transition to 1BCA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

B. Complete 1BEP ES-1.3; THEN transition to 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

C. Align the 1A CS train for recirculation AND transition to 1BCA-1.1 LOSS OF EMERGENCY COOLANT RECIRCULATION.

D. Complete 1BEP ES-1.3; THEN transition to 1BCA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

Answer	A

Answer Explanation

A - Correct: 1BCA-1.1 transition is directed by 1BEP ES-1.3 OAS after step 6, if ECCS recirculation is established and subsequently lost. Additionally, the OAS directs stopping any ECCS or CS pump if RWST level drop to 9% and its associated RWST suction valve is open. Isolation of ECCS pumps from RWST does not happen until steps 8 & 9 of 1BEP-ES1.3.

B - Plausible: Because these are actions in 1BEP ES-1.3 and normal procedure path.

C - Plausible: Because these are actions in 1BCA-1.1.

D - Plausible: Because these are actions in 1BEP ES-1.3 and Containment pressure is high, but only a YELLOW path CSF exists prior to stopping the 1A CS pump.

Question Information

Topic	Byron 2023 NRC Exam Q77				
User ID	RE1WE11-C17-01			**System ID**	2498520
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only			
Question Type	Bank from Bwd Cert exam bank	**Difficulty**	N/A
Technical Reference and Revision #	1BCA-1.1 Rev. 302, 1BEP ES-1.3 Rev. 305		
Training Objective	T.EP02-01-E		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring the examinee to interpret the significance of having an RH pump running during a large break LOCA.
SRO-Only Justification	The question is SRO only because it requires assessment of plant conditions and selection of emergency procedure for mitigation.
Additional Information	None

K/A Reference(s)

EPE.011.EA2.05	Safety Function	Tier 1	Group	RO Imp: 4.4	SRO Imp: 4.1
Ability to determine and/or interpret the following as they apply to a Large-Break LOCA: (CFR: 43.5 / 45.13) Significance of ECCS pump operation					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 78	ID: 2498524	Points: 1.00

- Unit 1 is at 20% power, ramping down at 5 MW/min per 1BGP 100-4, POWER DESCENSION.
- The 1A RCP had elevated but stable motor bearing temperatures.
- One minute ago, the 1A RCP motor bearing temperature began rising again.
- 1-7-A5 RCP 1A BRNG CC WTR FLOW LOW annunciator has alarmed.
- The crew has entered 1BOA PRI-6, COMPONENT COOLING MALFUNCTION.

1. Per 1BOA PRI-6, OAS, what is the motor bearing temperature at which 1A RCP must be tripped?

2. If the maximum motor bearing temperature is exceeded, what will the SRO direct?

A. 1. 195°F
2. Trip the RCP and shutdown the reactor per 1BGP 100-5, PLANT SHUTDOWN AND COOLDOWN while continuing with 1BOA RCP-2.

B. 1. 225°F
2. Trip the RCP and shutdown the reactor per 1BGP 100-5, PLANT SHUTDOWN AND COOLDOWN while continuing with 1BOA RCP-2.

C. 1. 195°F
2. Trip the Reactor, trip 1A RCP, and then enter 1BEP-0, REACTOR TRIP OR SAFETY INJECTION while continuing with 1BOA RCP-2.

D. 1. 225°F
2. Trip the Reactor, trip 1A RCP, and then enter 1BEP-0, REACTOR TRIP OR SAFETY INJECTION while continuing with 1BOA RCP-2.

Answer	C

Answer Explanation

A - Plausible: Because the reactor will not automatically trip on a single RCP trip with power below P-8. However, the plant safety analysis does not consider 3 loop operation in Mode 1. Additionally, Tech Spec 3.4.4 does not allow 3 loop operation. Therefore, continuing the shutdown into Mode 2 and using 1BGP 100-5 violates the plant safety analysis and TS 3.4.4 and is incorrect. Per BAR 1-13-A3, IF in MODE 1 or 2, and RCP breaker has opened or low flow conditions (<90%) exist, THEN VERIFY/INITIATE reactor trip and go to 1BEP-0, Reactor Trip or Safety Injection.

B - Plausible: Because per 1BOA PRI-6, an RCP must be tripped if pump lower bearing temperature exceeds 225°F. See answer A explanation for 2nd part plausibility.

C - Correct: 195°F is the maximum motor bearing temperature that an RCP is allowed to operate at per 1BOA PRI-6. The OAS directs trip the reactor, trip the affected RCP, perform 1BEP-0 while continuing with this procedure if RCP cooling can not be maintained.

D - Plausible: Because per 1BOA PRI-6, an RCP must be tripped if pump lower bearing temperature exceeds 225°F.

Question Information

Topic	Byron 2023 NRC Exam Q78				
User ID	SE10017-N02			**System ID**	2498524
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only			
Question Type	Modified from 2021 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA PRI-6 Rev. 114		
Training Objective	T.OA17-03		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring the examinee to determine when to trip an RCP on high bearing temperature.
SRO-Only Justification	The question is SRO only because it requires assessment of plant conditions and selection of emergency procedure for mitigation.
Additional Information	None

Original Question:

- Unit 1 is at 20% power, ramping down at 5 MW/min per 1BGP 100-4, POWER DESCENSION.
- The 1A RCP had elevated but stable lower bearing temperature.
- One minute ago the 1A RCP lower bearing temperature began rising again.
- 1-7-C2 RCP LOWER BRNG TEMP HIGH annunciator has alarmed.
- The crew has entered 1BOA RCP-2 LOSS OF SEAL COOLING UNIT 1.

1. Per 1BOA RCP-2 Step 2, MONITOR RCP TEMPERATURES, what is the lower bearing temperature at which 1A RCP must be tripped?

2. If the maximum lower bearing temperature is exceeded, what will the SRO direct?

A. 1. 195°F
2. Trip the RCP and shutdown the reactor per 1BGP 100-5, PLANT SHUTDOWN AND COOLDOWN while continuing with 1BOA RCP-2.

B. 1. 225°F
2. Trip the RCP and shutdown the reactor per 1BGP 100-5, PLANT SHUTDOWN AND COOLDOWN while continuing with 1BOA RCP-2.

C. 1. 195°F
2. Trip the Reactor, trip 1A RCP , and then enter 1BEP-0, REACTOR TRIP OR SAFETY INJECTION while continuing with 1BOA RCP-2.

D. 1. 225°F
2. Trip the Reactor, trip 1A RCP , and then enter 1BEP-0, REACTOR TRIP OR SAFETY INJECTION while continuing with 1BOA RCP-2.

Answer: D

K/A Reference(s)

APE.026.AA2.04	Safety Function	Tier 1	Group	RO Imp: 2.8	SRO Imp: 3.3
Ability to determine and/or interpret the following as they apply to Loss of Component Cooling Water: (CFR: 41.10 / 43.5 / 45.13) The normal values and upper limits for the temperatures of the components cooled by CCW					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 79	ID: 2498557	Points: 1.00

- Unit 1 was at 100% power when the following occurs:
- 1RY455A PZR PORV failed partially OPEN and could NOT be isolated (block valve will NOT close).
- The crew tripped the reactor, actuated SI and entered 1BEP-0, REACTOR TRIP OR SI.

Assume the following plant trends continue the entire time the crew is performing 1BEP-0:
- Pressurizer level is RISING.
- RCS pressure is DROPPING.
- PRT pressure is RISING.

With the above conditions, the next procedure transition will be from 1BEP-0, step _____(1)_____ to _____(2)_____.

A. (1) step 7, CHECK PZR PORVS AND SPRAY VALVES
(2) 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT

B. (1) step 14, CHECK IF RCS IS INTACT
(2) 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT

C. (1) step 15, CHECK IF ECCS FLOW SHOULD BE REDUCED
(2) 1BEP ES-1.1, SI TERMINATION

D. (1) step 21, CHECK IF RH PUMPS SHOULD BE STOPPED
(2) 1BEP ES-1.1, SI TERMINATION

Answer	A

Answer Explanation

A - Correct: Per 1BEP-0, step 7, if an un-isolable PZR Porv is not closed with pressure is below 2315 psig, the crew will transition to 1BEP-1.

B - Plausible: Because 1BEP-0, step 14 checks for elevated containment rads, pressure and floor drain sump level. Any would be a condition for transition to 1BEP-1. However, the crew should transition to 1BEP-1 at 1BEP-0, step 7.

C - Plausible: Because 1BEP-0, step 15 checks for RCS subcooling (acceptable), SG heat sink (available), PZR level (> 12%) and RCS pressure (stable or rising). If all conditions are met, the crew will transition to 1BEP ES-1.1. However, the crew should transition to 1BEP-1 at 1BEP-0, step 7.

D - Plausible: Because 1BEP-0, step 21 checks RCS pressure (>325 psig) and trend (stable or rising). If either is not met the crew should transition to 1BEP-1. However, the crew should transition to 1BEP-1 at 1BEP-0, step 7.

Question Information

Topic	Byron 2023 NRC Exam Q79				
User ID	SE10027-N02			**System ID**	2498557
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BEP-0 Rev. 306		
Training Objective	T.EP02-01-B		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring the examinee to recognize abnormal condition that is entry level for an emergency procedure.
SRO-Only Justification	The question is SRO only because it requires assessment of plant conditions and selection of emergency procedure for mitigation.
Additional Information	None

K/A Reference(s)

P2.4.04	Safety Function	Tier	Group	RO Imp: 4.5	SRO Imp: 4.7
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures (CFR: 41.10 / 43.2 / 45.6)					
G.APE.027	Safety Function	Tier 1	Group	RO Imp:	SRO Imp:
Pressurizer Pressure Control System Malfunction					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 80	ID: 2498560	Points: 1.00

During performance of 1BCA-0.0, LOSS OF ALL AC POWER, when shall the Shift manager evaluate entry into 10 CFR 50.54(x)?

A. Entry into a BFSG guideline is directed.

B. All 4kV ESF bus loads are placed in PULL OUT.

C. A Status Tree Red Path procedure entry criteria is met, but NOT implemented.

D. Entry into ATTACHMENT B, EXTENDED LOSS OF ALL AC POWER RESPONSE is required.

Answer	A

Answer Explanation

A - Correct: Per 1BCA-0.0 note prior to step 1, The Shift Manager shall evaluate entry into 10 CFR 50.54(x) and 10 CFR 73.55(p) when entry into a BFSG is directed.

B - Plausible: Because placing all 4kV ESF bus loads are placed in PULL OUT under normal circumstances would require multiple Tech Spec entries. However, because it is an action directed by an approved plant procedure, no 10 CFR 50.54(x) evaluation by the SM is required.

C - Plausible: Because Status Tree Red paths normally take precent over EPs. However, 1BCA-0.0 note prior to step 1, directs that Byron Status Trees should be monitored for information only. BFRs should NOT be implemented.

D - Plausible: Because entry into 1BCA-0.0, Attachment B is usually preceded by a BFSG implementation. However, the entry into Attachment B itself is an action directed by an approved plant procedure. Therefore, no 10 CFR 50.54(x) evaluation by the SM is required.

Question Information

Topic	Byron 2023 NRC Exam Q80				
User ID	SE10055-N01			System ID	2498560
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.1 Conditions and limitations in the facility license.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	Difficulty	N/A
Technical Reference and Revision #	1BCA-0.0 Rev. 305		
Training Objective	T.CA1-06		

Previous NRC Exams Use	None

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of conservative decision making by entering guides that depart from a license condition or a technical specification in an emergency when this action is immediately needed to protect the public health and safety.
SRO-Only Justification	The question is SRO only because it is information in a EP note that is the direct responsibility of the SRO/SM.
Additional Information	None

K/A Reference(s)

P2.1.39	Safety Function	Tier	Group	RO Imp: 3.6	SRO Imp: 4.3
Knowledge of conservative decision-making practices (CFR: 41.10 / 43.5 / 45.12)					
G.EPE.055	Safety Function	Tier 1	Group	RO Imp:	SRO Imp:
Station Blackout					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 81	ID: 2498564	Points: 1.00

- Unit 1 was ramping up from 80% to full power.
- ALL systems were normally aligned
- The NSO was performing intermittent MANUAL rod withdrawals per the reactivity plan.

The following occurs:
- Instrument bus 113 faults and is de-energized.
- The RO notes annunciator 1-10-B5, PWR RNG FLUX HIGH ROD STOP, is in alarm.

With the above conditions, MANUAL control rod WITHDRAWAL is ___(1)____ and the procedure steps the crew will perform to clear the annunciator 1-10-B5 alarm are in ____(2)_____.

A. (1) AVAILABLE
(2) 1BOA ELEC-2, LOSS OF INSTRUMENT BUS

B. (1) AVAILABLE
(2) 1BOA INST-1, NUCLEAR INSTRUMENTATION MALFUNCTION

C. (1) NOT available
(2) 1BOA ELEC-2, LOSS OF INSTRUMENT BUS

D. (1) NOT available
(2) 1BOA INST-1, NUCLEAR INSTRUMENTATION MALFUNCTION

Answer	D

Answer Explanation

A - Plausible: Because 1BOA ELEC-2 is the procedure entered for the loss of the instrument bus, but it kicks out to 1BOA INST-1 to clear the alarm and regain rod withdrawal capability. It's incorrect because manual control rod withdrawal will not be available. In addition, 1BOA INST-1 is the procedure that will be used to clear the alarm. On a loss of instrument bus 113, PRNI N-43 will also become de-energized. This will cause rods to NOT withdraw in Manual or AUTO because the High Flux rod stop (C-2) is a 1 out of 4 coincidence logic. It's plausible because other rod stop alarms such as OTDT rod stop (C-3) and OPDT rod stop (C-4) require a 2 out of 4 coincidence to effect a rod stop.

B - Plausible: Because other rod stop alarms such as OTDT rod stop (C-3) and OPDT rod stop (C-4) require a 2 out of 4 coincidence to effect a rod stop. 1BOA INST-1 is the procedure that will be used clear the alarm and regain rod withdrawal capability. It's incorrect because manual control rod withdrawal will not be available. On a loss of instrument bus 113, PRNI N-43 will also become de-energized. This will cause rods to NOT withdraw in Manual or AUTO because the High Flux rod stop (C-2) is a 1 out of 4 coincidence logic.

C - Plausible: Because 1BOA ELEC-2 is the procedure entered for the loss of the instrument bus, but it kicks out to 1BOA INST-1 to actually clear the alarm and regain rod withdrawal capability. It's incorrect because although manual control rod withdrawal will NOT be available, 1BOA INST-1 is the correct procedure that will be used to clear the alarm and regain rod withdrawal capability.

D - Correct: On a loss of instrument bus 113, PRNI N-43 will also become de-energized. This will cause rods to NOT withdraw in Manual or AUTO because the High Flux rod stop (C-2) is a 1 out of 4 coincidence logic. In addition, 1BOA INST-1 is the procedure that clears the alarm and allows the crew to regain rod withdrawal capability. The steps that clear the alarm are also found in BAR 1-10-B5.

Question Information

Topic	Byron 2023 NRC Exam Q81				
User ID	SE10057-N01			**System ID**	2498564
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only			
Question Type	Bank from 2022 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	BAR 1-10-B5 Rev. 2 1BOA INST-1 Rev. 110		
Training Objective	T.OA10-09		
Previous NRC Exams Use	2022 Byron NRC exam		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to verify alarm coincidence (availability of manual rod withdrawal) and operating controls in the alarm response procedure (rod stop bypass switch).
SRO-Only Justification	The question is SRO level because assessment of plant conditions and the selection of a procedure to recover from a malfunction. Specifically, it requires selection of procedure that has specific actions to bypass the alarm function that are not in the initial BOA procedure that would not be entered initially for the malfunction.
Additional Information	None

K/A Reference(s)

P2.4.50	Safety Function	Tier	Group	RO Imp: 4.2	SRO Imp: 4.0
Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure (CFR: 41.10 / 43.5 / 45.3)					
G.APE.057	Safety Function	Tier 1	Group	RO Imp:	SRO Imp:
Loss of Vital AC Electrical Instrument Bus					

Learning Objective(s)

2023 NRC SRO Exam

User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 82	ID: 2498589	Points: 1.00

- Unit 1 is at 75% Reactor power.
- A malfunction in the Rod Control circuitry caused a continuous rod bank withdrawal.
- Control rod motion was stopped by placing the ROD BANK SELECT SWITCH in SBD (Shutdown Bank D).
- Control Bank D stepped out 10 steps during the event.
- Tave rose to 2°F above Tref.

Per Tech Spec 3.1.4, Rod Group Alignment Limits, the control bank D rods are _____(1)_____ .

After the rod control circuitry malfunction is repaired, the SRO will direct the control rods to be returned to original position using ____(2)____.

A. (1) OPERABLE
(2) 1BOA ROD-1, UNCONTROLLED ROD MOTION

B. (1) OPERABLE
(2) 1BOA ROD-3, DROPPED OR MISALIGNED ROD

C. (1) INOPERABLE
(2) 1BOA ROD-1, UNCONTROLLED ROD MOTION

D. (1) INOPERABLE
(2) 1BOA ROD-3, DROPPED OR MISALIGNED ROD

Answer	A

Answer Explanation

A - Correct: Per Tech Spec 3.1.4 bases "The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e. trippable to meet SDM) are separate from the alignment requirements." Because there was no stem information of the rods being stuck (untrippable) the rods are considered operable. 1BOA ROD-1 contain the step to restore rods to the original position in step 7.

B - Plausible: Because 1BOA ROD-3 contains steps for recovering a dropped or misaligned rod. The stem scenario could easily be confused for misaligned rods (not in proper auto or manual position).

C - Plausible: Because control rods were not properly responding to plant conditions because of the rod control circuitry malfunction. However, the operability of the rods is not based upon the automatic ability to perform the control function, but rather whether they will perform the SDM function.

D - Plausible: Because control rods were not properly responding to plant conditions because of the rod control circuitry malfunction. However, the operability of the rods is not based upon the automatic ability to perform the control function, but rather whether they will perform the SDM function. Also, 1BOA ROD-3 contains steps for recovering a dropped or misaligned rod. The stem scenario could easily be confused for misaligned rods (not in proper auto or manual position).

Question Information

Topic	Byron 2023 NRC Exam Q82				
User ID	SE20001-N01			**System ID**	2498589
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only			
Question Type	New for 2023 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Tech Spec 3.1.4 Bases Rev. 59 1BOA ROD-1 Rev. 105		
Training Objective	T.OA32-06, T.OA32-02		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to make operational judgement based upon operational characteristics.
SRO-Only Justification	The question is SRO level because assessment of plant conditions and the selection of a procedure to recover from a malfunction. Specifically, it requires selection of procedure that has specific actions to realign control rods after a uncontrolled rod withdrawal.
Additional Information	None

K/A Reference(s)

P2.1.07	Safety Function	Tier	Group	RO Imp: 4.4	SRO Imp: 4.7
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (CFR: 41.5 / 43.5 / 45.12 / 45.13)					
G.APE.001	Safety Function	Tier 1	Group	RO Imp:	SRO Imp:
Continuous Rod Withdrawal					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 83	ID: 2498600	Points: 1.00

- BOTH Units are at 100% power.
- An 0WX01T, RADWASTE RELEASE TANK, liquid release is planned for later today.
- An EO reports 0PR10J, STATION BLOWDOWN RAD MONITOR, has the following local indications:
  - The red HIGH RADIATION light is NOT LIT.
  - The blue INTERLOCK light is LIT.
  - The green INSTRUMENT AVAILABLE light is NOT LIT.

Repair estimate is 5 days to fix the 0PR10J skid.
The Unit Supervisor will…

(REFERENCE PROVIDED)

A. allow the liquid release to occur without restriction.

B. estimate the CW blowdown flow rate once every four hours and allow the liquidrelease to occur.

C. notify Chemistry to perform grab sampling of the station blowdown line once every 12 hours and allow the liquid release to occur.

D. NOT allow the liquid release until two independent samples are analyzed and release rate and valve lineup are verified by two qualified staff members.

Answer	C

Answer Explanation

A - Plausible: Because 0PR010J does not have an interlock function. Therefore, the examinee may conclude that having 0PR01J operable is sufficient for an effluent release from 0WX01T.

B - Plausible: Because the examinee may confuse the required action for a faulted CW flow loop with an 0PR10J inoperability.

C - Correct: With the interlock light lit and the available and high radiation light not lit, 0PR10J is in an operate failure condition. 0PR10J does not have any automatic actuations. With 0PR10J inoperable, per 0BOL 11.a, LCOAR TABLE RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 12 hours, grab samples are collected and analyzed.

D - Plausible: Because the examinee may confuse the 0PR10J inoperability required actions with those of 0PR01J.

Question Information

Topic	Byron 2023 NRC Exam Q83				
User ID	SE20001-N01			**System ID**	2498600
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	REFERENCE PROVIDED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

NRC Exams Only			
Question Type	Bank from Bwd 2022 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	0BOL 11.a Rev. 13		
Training Objective	7E.TS-006-A		
Previous NRC Exams Use	None		

References Provided	0BOL 11.a Rev. 13
K/A Justification	Meets the K/A by requiring ability to determine plant configuration (whether or not to allow a release to continue) using plant control documentation.
SRO-Only Justification	The question is SRO level because it involves the application of Required Action Statements for identified equipment conditions affecting ODCM and RETS requirements.
Additional Information	None

K/A Reference(s)

P2.2.15	Safety Function	Tier	Group	RO Imp: 3.9	SRO Imp: 4.3
Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, lineups, or tagouts (reference potential) (CFR: 41.10 / 43.3 / 45.13)					
G.APE.059	Safety Function	Tier 1	Group	RO Imp:	SRO Imp:
Accidental Liquid Radwaste Release					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 84	ID: 2498603	Points: 1.00

- Unit 1 was in a refueling outage when Control Room received a report of a dropped fuel assembly.
- The Fuel Building Fuel Handling Incident Area monitors are reading;
  0RE-AR55 – 1115 mR/HR
  0RE-AR56 – 1075 mR/HR
- The Containment Fuel Handling Incident Area monitors are reading;
  1RE-AR011 – 50 mR/HR
  1RE-AR012 – 55 mR/HR

Per 1BOA REFUEL-1, Fuel Handling Emergency, the rad monitor indications are used to determine __(1)__ .
This event should be classified as an __(2)__ .

(REFERENCE PROVIDED)

A. (1) the location of the accident
   (2) Unusual Event

B. (1) the location of the accident
   (2) Alert

C. (1) whether or not the affected area must be evacuated
   (2) Unusual Event

D. (1) whether or not the affected area must be evacuated
   (2) Alert

Answer	B

Answer Explanation

A - Plausible: Because an Unusual Event also uses the rad monitors in Table R1 as a criterion for EAL classification.

B - Correct: 1BOA REFUEL-1 uses the incident monitors to determine the location of the accident. The affected area is automatically evacuated regardless of area rad monitor readings. RA2, criteria is MET for an Alert classification.

C - Plausible: Because ARM activity levels are used as criteria for specific operator actions. However, in 1BOA REFUEL-1, the ARMs are used as an entry condition (rising rad levels with no specific rad level criteria). Also, because an Unusual Event uses the rad monitors in Table R1 as a criterion for EAL classification.

D - Plausible: Because ARM activity levels are used as criteria for specific operator actions. However, in 1BOA REFUEL-1, the ARMs are used as an entry condition (rising rad levels with no specific rad level criteria).

Question Information

Topic	Byron 2023 NRC Exam Q84				
User ID	SE20061-N01			**System ID**	2498603
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	REFERENCE PROVIDED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.7 Fuel handling facilities and procedures.

NRC Exams Only			
Question Type	Bank from Byron 2019-2 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA REFUEL-1, Fuel Handling Emergency Rev. 110 EP-AA-1002 Addendum 3 Rev. 006		
Training Objective	7F.ZP-001-A-1 Rev, 006		
Previous NRC Exams Use	None		

References Provided	EP-AA-1002 Addendum 3, EMERGENCY ACTION LEVELS FOR BYRON STATION
K/A Justification	Meets the K/A by requiring ability to interpret ARM system alarms and how they are used in the AOP to direct actions.
SRO-Only Justification	This question is SRO only because requires E-plan classification.
Additional Information	None

K/A Reference(s)

APE.061.AA2.02	Safety Function	Tier 1	Group	RO Imp: 3.0	SRO Imp: 3.1
Ability to determine and/or interpret the following as they apply to Area Radiation Monitoring System Alarms: (CFR: 43.5 / 45.13) Normal operating characteristics					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 85	ID: 2498610	Points: 1.00

- An inadequate core cooling event is in progress on Unit 1.
- The crew is performing 1BFR-C.1, RESPONSE TO INADEQUATE CORE COOLING.

Currently, the crew is at step 16, CHECK IF RCPs SHOULD BE STARTED.
- ALL RCPs are shutdown and available.
- CETCs are 1235°F and slowly rising.
- Containment is ADVERSE.

The SRO desires to start an RCP with NORMAL SUPPORT CONDITIONS.

The following conditions exist:

RCS Loop	SG NR Levels	RCP No. 1 Seal DP
1A	36%	190 psid
1B	15%	230 psid
1C	34%	210 psid
1D	28%	220 psid

The SRO will direct the crew to start the...

A. 1A RCP

B. 1B RCP

C. 1C RCP

D. 1D RCP

Answer	C

Answer Explanation

A - Plausible: Because the 1A loop has the highest SG level. However, the RCP No.1 seal D/P is below the required support condition criteria.

B - Plausible: Because the 1B loop has the highest RCP No.1 seal D/P. However, SG level is below the required RCP available loop criteria (but would be adequate if containment was not adverse).

C - Correct: Per 1BCA FR-C.1, RCP No. seal D/P must be >200 psid for adequate support conditions (step 3). Also, SG NR level must be >10% (>31% adverse containment) for a RCS loop to be considered available (step 16). The note prior to step 16 directs that "Normal conditions are desired but NOT required for starting RCPs". This note applies to the RCP No.1 seal D/P. The SG NR levels must meet the available loop criteria, otherwise an RCP must not be started in that loop. The question stem stated that the SRO is going to start the RCP with adequate support conditions (adequate No.1 seal D/P). Therefore, the 1C loop is the only loop that meets both criteria.

D - Plausible: Because the 1D loop has adequate RCP No.1 seal D/P. However, SG level is below the required RCP available loop criteria (but it is higher than the 1B loop and would be adequate if containment was not adverse).

Question Information

Topic	Byron 2023 NRC Exam Q85				
User ID	SE1WE06-N01			**System ID**	2498610
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BFR-C.1 Rev. 301		
Training Objective	T.FR02-02		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to interpret RCP No. 1 seal D/P during an inadequate core cooling event.
SRO-Only Justification	Question is SRO level because the SRO it requires detailed knowledge of procedure steps not contained in an OAS.
Additional Information	None

K/A Reference(s)

W.E06.EA2.06	Safety Function	Tier	Group	RO Imp: 3.2	SRO Imp: 3.2
Ability to determine and/or interpret the following as they apply to Degraded Core Cooling: (CFR: 41.10 / 43.5 / 45.13) RCP No. 1 seal D/P					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 86	ID: 2501759	Points: 1.00

- Unit 1 is at 90% power.
- LTDWN HX OUT TEMP CONTROL 1CC130A, 1TK-0130, causes letdown temperature to lower to 75°F.

Assuming NO operator actions are taken, the VCT boron concentration will ___(1)___, AND the correct procedure the US will use to mitigate the event is ___(2)___.

A. (1) lower
(2) 1BOA PRI-12, UNCONTROLLED DILUTION.

B. (1) rise
(2) 1BOA PRI-6, COMPONENT COOLING MALFUNCTION.

C. (1) lower
(2) 1BOA PRI-6, COMPONENT COOLING MALFUNCTION.

D. (1) rise
(2) 1BOA ROD-1, UNCONTROLLED ROD MOTION.

Answer	A

Answer Explanation

A - Correct: 1BOA PRI-12, step 3, will check for low letdown temperature and then manually control letdown temperature or bypass the demins by placing 1CV129 to the VCT position. This will stop the dilution.

B - Plausible: Because letdown temperature effects on boron concentration are a common misunderstanding. The affected controller is a CC flow controller on the CVCS system and could plausibly be fixed utilizing 1BOA PRI-6. The hi letdown temperature alarm is checked in 1BOA PRI-6 step 6 to ensure proper temperature control.

C - Plausible: Because the boron concentration will drop and the affected controller is a CC flow controller on the CVCS system and could plausibly be fixed utilizing 1BOA PRI-6. The hi letdown temperature alarm is checked in 1BOA PRI-6 step 6 to ensure proper temperature control.

D - Plausible: Because letdown temperature effects on boron concentration are a common misunderstanding. The letdown temperature is checked greater than 80°F in 1BOA PRI-12 step 3 to ensure proper temperature control and will correct the temperature issue.

Question Information

Topic	Byron 2023 NRC Exam Q86				
User ID	SS20004-N02			**System ID**	2501759
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only			
Question Type	Bank from Bwd 2018 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA PRI-12 Rev. 106		
Training Objective	7D.OA-021-A		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to recognize abnormal CVCS system operating parameters that are entry-level conditions for abnormal operating procedures.
SRO-Only Justification	Question is SRO level because it requires assessment of plan conditions and selection of appropriate procedure to mitigate the consequences of the malfunction. The conditions are not entry criteria or immediate actions of either procedure and not part of the overall mitigative strategy.
Additional Information	None

K/A Reference(s)

P2.4.04	Safety Function	Tier	Group	RO Imp: 4.5	SRO Imp: 4.7
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures (CFR: 41.10 / 43.2 / 45.6)					
G.SYS.004	Safety Function	Tier 2	Group	RO Imp:	SRO Imp:
CVCS Chemical and Volume Control System					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 87	ID: 2498635	Points: 1.00

- Unit 1 is at 100% power.
- The 1A Containment Chiller and 1A Containment Chilled Water Pump are operating.
- An inadvertent Phase A Containment Isolation signal actuates on A Train ONLY.
- The crew enters 1BOA PRI-13, RECOVERY FROM INADVERTENT PHASE A CONTAINMENT ISOLATION.
- When the crew tried to reset the CNMT Isol Phase A signal. the MCR reset pushbutton did NOT work.

The 1A Containment Chilled Water Pump will ___(1)___ and the steps for LOCALLY resetting the CNMT Isol Phase A signal relays are found in _____(2)_____.

A. (1) TRIP
(2) 1BOA ELEC-5, LOCAL EMERGENCY CONTROL OF SAFE SHUTDOWN EQUIPMENT.

B. (1) TRIP
(2) 1BOA PRI-5, CONTROL ROOM INACCESSIBILITY.

C. (1) NOT trip
(2) 1BOA ELEC-5, LOCAL EMERGENCY CONTROL OF SAFE SHUTDOWN EQUIPMENT.

D. (1) NOT trip
(2) 1BOA PRI-5, CONTROL ROOM INACCESSIBILITY.

Answer	B

Answer Explanation

A - Plausible: Because 1BOA ELEC-5 has multiple attachments for local manipulation of plant equipment.

B - Correct: The 1A VP WO pump is interlocked to trip with 1 or more containment isolation valves closed in BOTH train A and train B flow paths. Because 1WO056B, Inside containment isolation valve (on the B train flow path) receives a train A phase A signal, the pump will trip. If a phase A isolation signal will not reset from the MCR pushbuttons, 1BOA PRI-13 directs a reset per 1BOA PRI-5, attachment E, step 7.

C - Plausible: Because it is a common misconception that the 1WO056A/B, Inside containment isolation valves receive phase A isolation signals from the opposite SSPS trains. 1BOA ELEC-5 has multiple attachments for local manipulation of plant equipment.

D - Plausible: Because it is a common misconception that the 1WO056A/B, Inside containment isolation valves receive phase A isolation signals from the opposite SSPS trains.

Question Information

Topic	Byron 2023 NRC Exam Q87				
User ID	SS10013-N01			**System ID**	2498635
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Electrical schematic 6E-1-4030WO03 Rev. F 1BOA PRI-13 Rev. 104 1BOA PRI-5 Rev. 111		
Training Objective	T.OA16-02		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to predict impacts of an inadvertent ESFAS and selecting procedure to mitigate the action.
SRO-Only Justification	Question is SRO level because it requires assessment of plan conditions and selection of appropriate procedure to mitigate the consequences of the malfunction.
Additional Information	None

K/A Reference(s)

SYS.013.A2.06	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp: 4.0
Ability to (a) predict the impacts of the following on the Engineered Safety Features Actuation System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 41.7 / 41.10 / 43.5 / 45.3 / 45.13) Inadvertent ESFAS actuation					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 88	ID: 2498770	Points: 1.00

- Unit 1 is operating at 100% with the 1A and 1B FW pumps in-service.
- The 1B CD/CB pump trips.
- The standby CD/CB pump FAILS to auto start (tripped immediately) and will not manually start.
- The following alarms came in:
  - 1-17-A9 CD/CB PUMP TRIP
  - 1-16-D1 FW PUMP SUCT HDR PRESS LOW

With the above conditions, the ____(1)____ will trip.

The crew enters 1BOA SEC-1, SECONDARY PUMP TRIP, and the SRO will direct a _____(2)_____ to INITIALLY mitigate this event.

A. (1) 1A FW Pump
(2) CD/FW RUNBACK

B. (1) 1A FW Pump
(2) REACTOR TRIP

C. (1) 1B FW Pump
(2) CD/FW RUNBACK

D. (1) 1B FW Pump
(2) REACTOR TRIP

Answer	A

Answer Explanation

A - Correct: Low FW Header suction pressure is a trip of the 1A FW pump only. 1BOA SEC-1, Attachments A and C both direct a turbine runback in these conditions.

B - Plausible: Because 1BOA SEC-1 will direct a reactor trip if no FW pumps are running in Attachment A or if less than two CD/CB pumps are running in Attachment C. However, these conditions are not met.

C - Plausible: Because the 1B FW pump was running during the event and a common misconception is the turbine driven FW pumps trip on low FW pump suction pressure.

D - Plausible: Because the 1B FW pump was running during the event and a common misconception is the turbine driven FW pumps trip on low FW pump suction pressure. Also, because 1BOA SEC-1 will direct a reactor trip if no FW pumps are running in Attachment A or if less than two CD/CB pumps are running in Attachment C. However, these conditions are not met.

Question Information

Topic	Byron 2023 NRC Exam Q88				
User ID	SS10059-N01			**System ID**	2498770
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only			
Question Type	Modified from Bwd 2022 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA SEC-1 Rev. 113 BAR 1-16-D1 Rev. 4		
Training Objective	T.OA36-05		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to predict impacts of a loss of condensate flow on the FW system and select the appropriate recovery procedure.
SRO-Only Justification	Question is SRO level because it requires assessment of plan conditions and detailed procedure knowledge not found in the OAS to mitigate the consequences of the malfunction.
Additional Information	None

Original question:

Unit 1 is operating at 100% with the 1A and 1B FW pumps in-service.

The 1B condensate/condensate booster pump trips.

The following alarms come in:

1-17-A9 CD/CB PUMP TRIP
1-17-B11 CB PUMP DSCH FLOW HIGH
1-17-D3 HD PUMP DSCH FLOW HIGH
1-17-D10 CD/CB PUMP LUBE OIL PP AUTO START
1-17-E2 HD PUMP GLND WTR INLET PRESS HIGH LOW
1-17-E9 CB PUMP GLND WTR INLET PRESS HIGH LOW
1-16-E1 FEEDWATER NPSH LOW

Before any operator actions are taken,

The 1A FW pump trips.

The following additional alarms come in:

1-16-A1 FW PUMP 1A TRIP

1-16-D2 FW PUMP DSCH FLOW HIGH

Which of the following automatic actions failed to occur?

The US will prioritize directing actions in accordance with 1BwOA SEC-1, SECONDARY PUMP TRIP, _____ to INITIALLY mitigate this event.

(1) The standby FW pump failed to auto-start.
(2) ATTACHMENT A, FW PUMP TRIP ≥ 15% POWER.

(1) The standby FW pump failed to auto-start.
(2) ATTACHMENT C, CD/CB PUMP TRIP MODE 1 OR 2.

(1) The standby CD/CB failed to auto-start.
(2) ATTACHMENT A, FW PUMP TRIP ≥ 15% POWER.

D. (1) The standby CD/CB failed to auto-start.
(2) ATTACHMENT C, CD/CB PUMP TRIP MODE 1 OR 2.

ANSWER C

K/A Reference(s)

SYS.059.A2.13	Safety Function	Tier 2	Group	RO Imp: 3.0	SRO Imp: 2.8
Ability to (a) predict the impacts of the following on the Main Feedwater System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) Loss of condensate/heater draining flow					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 89	ID: 2498784	Points: 1.00

- Both units are operating at 100%.
- The U-0 CC Heat Exchanger and Pump are mechanically and electrically aligned to Unit 1.

- Subsequently, abnormal CC surge tank level causes the crew to enter 1BOA PRI-6 COMPONENT COOLING MALFUNCTION.
- The crew determines that the U-0 CC Heat Exchanger has a large tube leak.

With the above conditions, _____(1)_____.

When the crew isolates the U-0 CC Heat Exchanger, LCO 3.7.7, CC System, entry will be required for _____(2)_____ .

A. (1) SX water is leaking INTO the CC system.
(2) Unit 1 ONLY

B. (1) SX water is leaking INTO the CC system.
(2) BOTH Units

C. (1) CC water is leaking INTO the SX system.
(2) Unit 1 ONLY

D. (1) CC water is leaking INTO the SX system.
(2) BOTH Units

Answer	D

Answer Explanation

A - Plausible: Because both CC and SX have other applications in the plant where CC would be the lower pressure system in a heat exchanger or SX would be the higher pressure system in a heat exchanger. Also, the U-0 heat exchanger was aligned to Unit 1 in the question stem. Therefore, a Unit 1 only LCO entry is plausible.

B - Plausible: Because both CC and SX have other applications in the plant where CC would be the lower pressure system in a heat exchanger or SX would be the higher pressure system in a heat exchanger.

C - Plausible: Because the U-0 heat exchanger was aligned to Unit 1 in the question stem. Therefore, a Unit 1 only LCO entry is plausible.

D - Correct: SX pumps operate at a discharge pressure of 90-100 psig. while CC pumps operate at a discharge pressure of 130-140 psig. Therefore, the CC water will be flowing into the SX system. This is confirmed by 1BOA PRI-6, Attachment B, Step 2, which list the CC Hx(s) as a possible location of CC system outleakage. Tech Spec 3.7.7 bases requires the U-0 CC Heat Exchanger as a CC system flowpath that is credited to both units. Therefore, both units must enter the LCO when the U-0 CC Heat Exchanger is isolated.

Question Information

Topic	Byron 2023 NRC Exam Q89				
User ID	SS20076-N03			**System ID**	2498784
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BOA PRI-6 Rev. 114 Tech Spec bases B 3.7.7 Rev. 0		
Training Objective	T.GP01-11		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to predict impacts on SX system of a heat exchanger load failure and use procedures to mitigate the consequences.
SRO-Only Justification	Question is SRO level because it requires knowledge of Tech Spec bases.
Additional Information	None

K/A Reference(s)

P2.2.25	Safety Function	Tier 3	Group	RO Imp: N/A	SRO Imp: 4.2
Knowledge of the bases in TS for limiting conditions for operation and safety limits (SRO Only) (CFR: 43.2)					
G.SYS.076	Safety Function	Tier	Group	RO Imp:	SRO Imp:
SW Service Water System					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 90	ID: 2501241	Points: 1.00

The 1PI-PC004, Containment Pressure (Wide Range) instrument on 1PM06J is suspect for failing its channel check. (see below)

Which Tech Spec is applicable to 1PI-PC004 that the SRO will evaluate for entry conditions?

A. 3.3.2 Engineered Safety Feature Actuation System Instrumentation

B. 3.3.3 Post Accident Monitoring Instrumentation

C. 3.3.4 Remote Shutdown System

D. 3.6.4 Containment Pressure

Answer	B

Answer Explanation

A - Plausible: Because these are the Hi containment pressure SI and CS actuation pressure instrumentation (1CS-934.935,936,937) which are Tech Spec 3.3.2 instruments.

B - Correct: The 1PI-PC004 & 1PI-PC005 are the containment wide range pressure indicators that are designated Tech Spec 3.3.3 Accident Monitoring Instrumentation. The table of applicable instrumentation in TS 3.3.3 contains Containment Pressure (Wide Range) and 1BOSR 3.3.1-1 specifically lists 1PI-PC-004 and 005.

C - Correct: Because Tech Spec 3.3.4 contains instrumentation that is applicable to channels on the RSP. Several of the RSP instrumentation are meters that are fed from the same transmitters as the MCR instrumentation. However, containment pressure is not on the RSP.

D - Plausible: Because 1PDI-VP236 has containment pressure indication displays at two locations in the MCR that are used for evaluating Tech Spec 3.6.4, Containment Pressure.

Question Information

Topic	Byron 2023 NRC Exam Q90				
User ID	SS20076-N03			**System ID**	2501241
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Tech Spec 3.3.3, Amendment 143 1BOSR 3.3.1-1 Rev. 23		
Training Objective	T.GP01-11		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to identify post accident instrumentation.
SRO-Only Justification	Question is SRO level because it requires knowledge of Tech Spec instrumentation that is not listed in the LCO or applicability statements (information that is below the double line).
Additional Information	None

K/A Reference(s)

P2.4.03	Safety Function	Tier	Group	RO Imp: 3.7	SRO Imp: 3.9
Ability to identify post-accident instrumentation (CFR: 41.6 / 45.4)					
G.SYS.103	Safety Function	Tier 2	Group	RO Imp:	SRO Imp:
CNT Containment System					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 91	ID: 2498838	Points: 1.00

- Unit 1 is at 100% power, preparing for a Tech Spec 3.0.3 action ramp.
- The RO has just completed an initial boration (per the pre-prepared REMA) and is waiting for observable RCS temperature affects.
- The RO was about to initiate the turbine ramp when the following occurs:

A PZR level control malfunction causes 1CV121, CENT CHG PUMPS FLOW CONTROL VALVE to throttle closed to 52 gpm.

Per OP-AA-300, REACTIVITY MANAGEMENT, the SRO will direct the turbine ramp to ...

A. be immediately INITIATED at the REMA directed ramp rate. Then, subsequently isolate letdown.

B. be immediately INITIATED at the REMA directed ramp rate. Then, use 1CV8104, EMER BORATION VALVE, for subsequent borations.

C. be immediately INITIATED at a SLOWER ramp rate than the REMA directed.

D. NOT be initiated until the SRO has evaluated the transient's affect on the ramp.

Answer	D

Answer Explanation

A - Plausible: Because a Tech Spec 3.0.3 action ramp is time sensitive and designed to meet the TS action Mode 3 requirements. However, there typically is margin in the REMA such that the ramp rate can be adjusted to accommodate delays. Also, because isolating letdown will be required if normal charging flow is not promptly restored.

B - Plausible: Because a Tech Spec 3.0.3 action ramp is time sensitive and designed to meet the TS action Mode 3 requirements. However, there typically is margin in the REMA such that the ramp rate can be adjusted to accommodate delays. Also, because the loss of charging flow is going to delay the rate of the initial boration. The emergency boration valve is used when the normal boration flowpath is not adequate. However, use of the emergency boration valve is limited to situations applicable to 1BOA PRI-2 and not appropriate for this scenario.

C - Plausible: Because the loss of charging flow would slow the rate of boron concentration change, if the change was a dilution into the VCT (vs. a boration). Therefore, it is logical that a slower ramp rate would be appropriate to match a misconception that the boron injection rate into the RCS is slower. However, prepared REMAs for planned reactivity events are typically organized with step reactivity additions (boron additions and rod insertions) to match the turbine load ramp rate. A slower ramp rate is not a solution after the transient is evaluated.

D - Correct: Per OP-AA-300, section 4.9.12, If a plant transient should occur during a planned reactivity manipulation (even if the two events are not related), then the reactivity manipulation should be STOPPED (place plant in a safe condition) until the SRO has EVALUATED that continuation of the reactivity manipulation is appropriate. Ramping the turbine is a significant reactivity manipulation that should be evaluated in this condition.

Question Information

Topic	Byron 2023 NRC Exam Q91				
User ID	SS20011-N01			**System ID**	2498838
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	New for Byron 2023 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	OP-AA-300 Rev. 14		
Training Objective	8E.AM-003		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of reactivity management procedures during a pressurizer level control malfunction event.
SRO-Only Justification	Question is SRO level because it is the SRO responsibility to direct the control room reactivity activities.
Additional Information	None

K/A Reference(s)

P2.1.37	Safety Function	Tier	Group	RO Imp: 4.3	SRO Imp: 4.6
Knowledge of procedures, guidelines, or limitations associated with reactivity management (CFR: 41.1 / 41.5 / 41.10 / 43.6 / 45.6)					
G.SYS.011	Safety Function	Tier 2	Group	RO Imp:	SRO Imp:
PZR LCS Pressurizer Level Control System					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 92	ID: 2498909	Points: 1.00

- A SGTR has occurred in the 1A S/G.
- The crew is depressurizing the RCS per 1BEP ES-3.3, POST-SGTR COOLDOWN USING STEAM DUMP, step 14, DEPRESSURIZE RCS TO MINIMIZE RCS-TO-SECONDARY LEAKAGE.
- The 1C Main Steamline rad monitors have recently gone into the alert/alarm condition.
- 1C S/G NR level is 43% and slowly rising.
- 1B and 1D S/G NR levels are at 35% and stable.
- CETCs are 400 ºF.
- RCS pressure is 600 psig.
- PZR level is 54% and slowly dropping.

The SRO will...

A. continue in 1BEP ES-3.3 and isolate AF flow to the 1C S/G.

B. continue in 1BEP ES-3.3 and concurrently enter 1BOA SEC-8, S/G TUBE LEAK.

C. transition to 1BEP-3, STEAM GENERATOR TUBE RUPTURE, after stabilizing the plant.

D. transition to 1BCA-3.1, SGTR WITH LOSS OF REACTOR COOLANT SUBCOOLED RECOVERY DESIRED.

Answer	C

Answer Explanation

A - Plausible: Because this would be correct if steamline rad monitors were not alarming.

B - Plausible: Because this is a valid distractor if the candidate does not recognize the multiple ruptured SG conditions and fails to implement the steps in 1BEP ES-3.3 OAS for multiple ruptured SG conditions.

C - Correct: Per the 1BEP ES-3.3 OAS, Stabilize the plant and return to 1BEP-3 step 1, if: Any intact SG level rises in an uncontrolled manner or any intact SG has abnormal secondary radiation.

D - Plausible: Because this is in the 1BEP ES-3.3 OAS summary to transfer to if subcooling is lost.

Question Information

Topic	Byron 2023 NRC Exam Q92				
User ID	SS20035-N02			System ID	2498909
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

NRC Exams Only

Question Type	Bank from Bwd 2020 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	1BEP ES-3.3, Rev. 301		
Training Objective	T.EP04-07		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to predict impact (determine that a SGTR is in progress on the 1C SG is implied by choosing the correct procedure transition) and use procedures to mitigate.
SRO-Only Justification	Question is SRO level because in requires assessment of plant conditions and selection of a procedure with which to proceed.
Additional Information	None

K/A Reference(s)

SYS.035.A2.01	Safety Function	Tier 2	Group	RO Imp: 4.6	SRO Imp: 4.4
Ability to (a) predict the impacts of the following on the Steam Generator System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5) Faulted, leaking, or ruptured S/G					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 93	ID: 2498948	Points: 1.00

- Unit 1 was shutdown 5 DAYS AGO and is currently in Mode 6.
- Fuel moves are in progress inside containment.
- 1AR011J, Containment Fuel Handling Incident Radiation Monitor detector fails.
- The SRO is evaluating the failure for Tech Spec entry requirements.

With the above conditions, the 1AR011J, Containment Fuel Handling Incident Radiation Monitor, actuation function __(1)__ required to be OPERABLE per Tech Specs because _____(2)_____.

A. (1) IS
(2) the potential for radioactive releases in this condition may EXCEED post-accident offsite doses being maintained within the limits of 10CFR50.67.

B. (1) IS
(2) each penetration providing access from the containment atmosphere to the outside atmosphere MAY be open and operator action CANNOT ensure post-accident offsite doses are maintained within the limits of 10CFR50.67.

C. (1) is NOT
(2) each penetration providing access from the containment atmosphere to the outside atmosphere MUST be closed by a manual or automatic isolation valve or blind flange.

D. (1) is NOT
(2) the potential for radioactive releases is minimized and operator action is sufficient to ensure post-accident offsite doses are maintained within the limits of 10CFR50.67.

Answer	D

Answer Explanation

A - Plausible: Because the containment isolation functions of 1AR011/012J must be operable when recently irradiated fuel moves are in progress. However, the definition of recently irradiated fuel is, fuel that has occupied part of a critical reactor core within the previous 48 hours. A reactor that has been shutdown for 5 days will not contain recently irradiated fuel. Part 2 is plausible because it is opposite the correct response and supports the distractor A answer to part 1.

B - Plausible: Because the containment isolation functions of 1AR011/012J must be operable when recently irradiated fuel moves are in progress. However, the definition of recently irradiated fuel is, fuel that has occupied part of a critical reactor core within the previous 48 hours. A reactor that has been shutdown for 5 days will not contain recently irradiated fuel. The part 2 distractor is plausible because, TS 3.3.6 allows recently irradiated fuel moves (with 1AR011/012J inoperable) if each penetration providing access from the containment atmosphere to the outside atmosphere is closed by a manual or automatic isolation valve, blind flange, or equivalent.

C - Plausible: Because TS 3.3.6 allows recently irradiated fuel moves (with 1AR011/012J inoperable) if each penetration providing access from the containment atmosphere to the outside atmosphere is closed by a manual or automatic isolation valve, blind flange, or equivalent. However, when moving fuel that is not recently irradiated, containment isolation is not required.

D - Correct: Per LCO 3.3.6 bases, While in MODES 5 and 6 without fuel handling in progress or when moving fuel in containment that is not RECENTLY IRRADIATED FUEL, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post-accident offsite doses are maintained within the limits of 10CFR50.67.

Question Information

Topic	Byron 2023 NRC Exam Q93				
User ID	SS20034-N03			**System ID**	2498948
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

NRC Exams Only			
Question Type	Bank from Bwd 2016 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	B 3.3.6 Rev. 89, page B 3.3.6-4		
Training Objective	T.GP01-11		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring ability to predict impact of an ARM malfunction and use procedures to mitigate the mitigate the consequences (applying TS 3.3.6 bases).
SRO-Only Justification	Question is SRO level because it requires knowledge of Tech Spec bases.
Additional Information	None

K/A Reference(s)

SYS.072.A2.01	Safety Function	Tier 2	Group	RO Imp: 3.3	SRO Imp: 3.3
Ability to (a) predict the impacts of the following on the Area Radiation Monitoring System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 43.3 / 45.13 / 9) ARM component failures					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 94	ID: 2498963	Points: 1.00

- Unit 1 is in MODE 6.
- Fuel load is in progress following a complete core offload.

Which of the following requires IMMEDIATE suspension of core alterations?

A. RCS temperature changes from 107 °F to 97 °F in 30 minutes due to an operator controlled evolution.

B. The last 2 RCS boron samples were 2340 ppm and 2315 ppm respectively.

C. ALL SR channels change from 11 cps to 20 cps following the insertion of the 11th fuel assembly into the core.

D. SR channel N-31 changes from 5 cps to 14 cps following the insertion of the 10th fuel assembly into the core. N-32 rose from 6 cps to 8 cps.

Answer	B

Answer Explanation

A - Plausible: Because OU-AP-200, section 3.1.1.14 directs: The Control Room and Operating Shift personnel are kept informed of any unusual plant conditions pertaining to refueling activities in progress (including temperature). However, because the evolution is controlled by operations and there is no specific guidance for stopping core alts due to temperature changes of the RCS, this answer is incorrect.

B - Correct: Per OU-AP-200 section 4.8.9 core alterations must be immediately suspended if boron concentration drops more than 20 ppm as determined by two successive samples.

C - Plausible: Because an unanticipated rise by factor of 2 on ALL responding NIs during any single load step after the first 8 fuel assemblies are loaded is criteria for suspending core alts. However, the distractor does not meet this criteria.

D - Plausible: Because an unanticipated rise by a factor of five on any individual NI after the initial 8 fuel assemblies are loaded is criteria for suspending core alts. However, the distractor does not meet this criteria.

Question Information

Topic	Byron 2023 NRC Exam Q94				
User ID	SS20034-N03			System ID	2498963
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.7 Fuel handling facilities and procedures.

NRC Exams Only			
Question Type	Bank from Bwd Cert exam bank	Difficulty	N/A

Technical Reference and Revision #	OU-AP-200 rev. 24
Training Objective	T.AM12-10
Previous NRC Exams Use	None

References Provided	None
K/A Justification	Meets the K/A by requiring knowledge of refueling administrative requirements.
SRO-Only Justification	Question is SRO level because it applies to unique SRO responsibilities during refueling.
Additional Information	None

K/A Reference(s)

P2.1.40	Safety Function	Tier 3	Group	RO Imp: 2.8	SRO Imp: 3.9
Knowledge of refueling administrative requirements (CFR: 41.10 / 43.5 / 43.6 / 45.13)					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 95	ID: 2498978	Points: 1.00

- Unit 1 is at 95% power.
- One Control Bank D DRPI has just failed.
- A plant transient occurs which requires a rapid power reduction to 80% power.
- Control Bank D control rods inserted 28 steps.

To comply with Technical Specification 3.1.7, ROD POSITION INDICATION, the Unit Supervisor will initiate actions to verify the position of the rods with failed DRPI __(1)__.

To determine the failed DRPI control rod position, Unit 1 will use PDMS, if OPERABLE, OR perform a __(2)__.

(REFERENCE PROVIDED)

A. (1) immediately
(2) flux map

B. (1) within 8 hours
(2) QPTR surveillance

C. (1) within 8 hours
(2) flux map

D. (1) immediately
(2) QPTR surveillance

Answer	A

Answer Explanation

A - Correct: Tech Spec 3.1.7, COND C is applicable and the required action is to immediately initiate action to verify the position of the rods with inoperable DRPIs. For the second part, the Bases for Tech Spec 3.1.7 states that either in-core detectors or PDMS can be used to determine rod position for a rod with a failed DRPI. In-core detectors are used to perform flux maps.

B - Plausible: Because the examinee may confuse the 8-hour completion time of COND A with the immediate completion time of the applicable COND C. The second part is plausible because a common misconception is based on required actions for a misaligned rod (TS 3.1.4) instead of the required actions for a failed DRPI.

C - Plausible: Because the examinee may confuse the 8-hour completion time of COND A with the immediate completion time of the applicable COND C.

D - Plausible: Because a common misconception is based on required actions for a misaligned rod (TS 3.1.4) instead of the required actions for a failed DRPI.

Question Information

Topic	Byron 2023 NRC Exam Q95				
User ID	SG10043-N01			**System ID**	2498978
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	REFERENCE PROVIDED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

NRC Exams Only			
Question Type	Bank from 2020 Bwd NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	Tech Spec 3.1.7 Amendment 202 and B 3.1.7 Rev. 109		
Training Objective	S.TS1-14		
Previous NRC Exams Use	None		

References Provided	TS 3.1.7, Rod Position Indication
K/A Justification	Meets the K/A by requiring ability to use procedures to determine the effects on reactivity of plant changes.
SRO-Only Justification	Question is SRO level because it requires examinee demonstrate knowledge of TS bases that is required to analyze TS required actions and terminology.
Additional Information	None

K/A Reference(s)

P2.1.43	Safety Function	Tier 3	Group	RO Imp: 4.1	SRO Imp: 4.3
Ability to use an online power distribution monitoring system and/or procedures to determine the effects on reactivity of plant changes, such as RCS temperature, secondary plant, or fuel depletion (CFR: 41.10 / 43.6 / 45.6)					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 96	ID: 2498989	Points: 1.00

Per CC-AA-103-100, CONFIGURATION CHANGE CONTROL FOR PERMANENT PHYSICAL PLANT CHANGES, when does "Operations Acceptance" of a permanent plant configuration change occur?

A. When Operations allows work to begin on the Work Order that implements the change.

B. When the Operations training is complete on the plant configuration change.

C. When the Engineering Change Package is approved for implementation.

D. When Operations agrees that the affected SSC(s) can be placed in service.

Answer	D

Answer Explanation

A - Plausible: Because this is a step in the configuration change implementation process.

B - Plausible: Because this is a step in the configuration change implementation process.

C - Plausible: Because this is a step in the configuration change implementation process.

D - Correct: CC-AA -103-100, section 2.16 Operations acceptance occurs when operations has reviewed the implementation status of the change and agrees that the change status is acceptable for subsequently placing the SSC into service.

Question Information

Topic	Byron 2023 NRC Exam Q96				
User ID	SG10043-N01			System ID	2498989
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 43.3 Facility licensee procedures required to obtain authority for design and operating changes in the facility.

NRC Exams Only			
Question Type	Bank from 2011 Bwd NRC exam	Difficulty	N/A
Technical Reference and Revision #	CC-AA-103-100 Rev. 2		
Training Objective	8E.AM-003		
Previous NRC Exams Use	None		

References Provided	None

K/A Justification	Meets the K/A by requiring examinee knowledge of process for making operating changes to the facility.
SRO-Only Justification	Question is SRO level because it is the SRO responsibility to review and accept implemented design changes.
Additional Information	None

K/A Reference(s)

P2.2.05	Safety Function	Tier 3	Group	RO Imp: 2.2	SRO Imp: 3.2
Knowledge of the process for making design or operating changes to the facility, such as 10 CFR 50.59, “Changes, Tests and Experiments,” screening and evaluation processes, administrative processes for temporary modifications, disabling annunciators, or installation of temporary equipment (CFR: 41.10 /43.3 / 45.13)					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 97 ID: 2498993 Points: 1.00

A work order requires a tagout that will take credit for administrative control of a plant condition to control the hazards for the planned work.

Per OP-AA-109-101, PERSONNEL AND EQUIPMENT TAGOUT PROCESS, the tagout will be designated as _____(1)_____ . In addition to the RO turnover sheet, the Tagout MUST also be tracked on the _____(2)_____ turnover sheet.

A. (1) Administrative
(2) Unit Supervisor

B. (1) Administrative
(2) Shift Manager

C. (1) Condition Dependent
(2) Unit Supervisor

D. (1) Condition Dependent
(2) Shift Manager

Answer	**C**

Answer Explanation

A – Plausible: An administrative tagout is a tagout that provides administrative control. However, no work is allowed under it.

B – Plausible: An administrative tagout is a tagout that provides administrative control. However, no work is allowed under it. OP-AA-109-101 specifically requires the US turnover to track condition dependent tagouts, not the SM turnover.

C – Correct: Per OP-AA-109-101, section 3.11, a Condition Dependent Tagout is a tagout that will take credit for administrative control of a plant condition to control the hazards for the planned work. Also, section 7.12.2.4 requires that Condition Dependent tagouts must be tracked on the US turnover sheet.

D – Plausible: OP-AA-109-101 specifically requires the US turnover to track condition dependent tagouts, not the SM turnover.

Question Information

Topic	Byron 2023 NRC Exam Q97				
User ID	SG20019-N02			**System ID**	2498993
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from 2020 Bwd NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	OP-AA-109-101, Rev. 18		
Training Objective	T.AM33-02		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring examinee knowledge of process for controlling equipment status.
SRO-Only Justification	Question is SRO level because it is specific knowledge of the SRO duties within the WTO process.
Additional Information	None

K/A Reference(s)

P2.2.14	Safety Function	Tier 3	Group	RO Imp: 3.9	SRO Imp: 4.3
Knowledge of the process for controlling equipment configuration or status (CFR: 41.10 / 43.3 / 45.13)					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 98	ID: 2498994	Points: 1.00

- A Unit 1 containment release package is being prepared per BCP 400-ECNMT/ROUTINE, GASEOUS EFFLUENT RELEASE FORM, ROUTINE CONTAINMENT RELEASE.
- Containment Noble Gas and Tritium activity samples were obtained on 10/1/23 at 1200 hours.
- ALL applicable Radiation Monitors are operable for the release.
- The SRO is reviewing the release for approval.

What should be listed as the Release Form expiration time, if any?

A. 10/2/23 at 0000 hours

B. 10/2/23 at 1200 hours

C. 10/4/23 at 1200 hours

D. N/A (no expiration time)

Answer	D

Answer Explanation

A – Plausible: Because 12 hours is common for Tech Spec and surveillance windows.

B – Plausible: Because 24 hours is common for Tech Spec and surveillance windows

C – Plausible: Because per BCP 400-TCNMT/ROUTINE, section 2.3.8.3, if 1RE-PR011 was out of service, the release form would expire 48 hours from the sample time.

D – Correct: Per BCP 400-TCNMT/ROUTINE, section 1.2.2, If the 1(2)RE-PR011 is in service then MARK the expiration time / date as “N/A”. As long as the applicable monitors 1(2)RE-PR001J and 1(2)RE-PR011J remain operable there is no expiration time for the containment release.

Question Information

Topic	Byron 2023 NRC Exam Q98				
User ID	SG30006-N03			System ID	2498994
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only

Question Type	Modified from 2022 Byron NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	BCP 400-ECNMT/ROUTINE, Rev. 2		
Training Objective	7E.AM-126-A		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring examinee ability to properly review radioactive release form for approval.
SRO-Only Justification	Question is SRO level because it is the SROs responsibility to review and approve the release prior the NSO initiating.
Additional Information	None

Original question:

A Unit 1 containment release package is being prepared per BCP 400-TCNMT/ROUTINE, GASEOUS EFFLUENT RELEASE FORM, ROUTINE CONTAINMENT RELEASE.
Containment Noble Gas and Tritium activity samples were obtained.
1RE-PR011 is OUT OF SERVICE.
The SRO is reviewing the release for approval.

Which of the following is the LONGEST time the SRO has, from sample time, to approve the release and give it to the NSO for initiating?

A. 12 hours

B. 24 hours

C. 48 hours

D. 72 hours

Answer	**C**

K/A Reference(s)

P2.3.06	Safety Function	Tier 3	Group	RO Imp: 2.0	SRO Imp: 3.8
Ability to approve liquid or gaseous release permits (CFR: 41.13 / 43.4 / 45.10)					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 99	ID: 2498995	Points: 1.00

- T.S. 3.0.3 was entered at 0320 hrs.
- At 0400 hrs, a plant shutdown was INITIATED.
- At 0900 hrs, the shutdown was stopped, while still in Mode 1, because T.S. 3.0.3 was exited.
- The plant then returned to 100% power.

Select the appropriate required LATEST notification time limit, if any, to the NRC.
This event is...

(REFERENCE PROVIDED)

A. not reportable.

B. reportable by 0720 hrs.

C. reportable by 0800 hrs.

D. reportable by 1300 hrs.

Answer	C

Answer Explanation

A – Plausible: Because the shutdown was not completed. SAF 1.2 has a requirement to submit a LER to the NRC within 60 days if the shutdown is completed. Because the shutdown was not completed, it is plausible that a candidate may select no reportability requirements.

B – Plausible: Because this is 4 hours from when TS 3.0.3 was entered.

C – Correct: Per SAF 1.2, Notify the NRC Operations Center via the ENS as soon as practical and in all cases, within four hours of the occurrence of the initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

D – Plausible: Because this is 4 hours from when the ramp was stopped.

Question Information

Topic	Byron 2023 NRC Exam Q99				
User ID	BYLC8EAM102001			**System ID**	2498995
Status	Active	**Point Value**	1.00	**Time (min)**	0

Cross Reference Number	REFERENCE PROVIDED
Cognitive Level	SRO-HIGH
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only			
Question Type	Bank from Byron LOR exam bank	**Difficulty**	N/A

Technical Reference and Revision #	LS-AA-1110 Rev. 33 Constellation Reportability Manual SAF LS-AA-1020 Rev. 34 Reportability Tables and Decision Trees
Training Objective	7E.AM-102-A
Previous NRC Exams Use	None

References Provided	LS-AA-1110, LS-AA-1120, LS-AA-1130, LS-MW-1310, LS-MW1320, LS-MW-1340, LLS-AA-1020, LS-AA-1010, collectively called the Constellation Reportability Manual.
K/A Justification	Meets the K/A by requiring examinee knowledge of events related to operational status that may be reportable to the NRC.
SRO-Only Justification	Question is SRO level because it is the SROs responsibility to review and initiate the reportability.
Additional Information	None

K/A Reference(s)

P2.4.30	Safety Function	Tier 3	Group	RO Imp: 2.7	SRO Imp: 4.1
Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator (CFR: 41.10 / 43.5 / 45.11)					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None

Question 100	ID: 2498998	Points: 1.00

· Byron Station has declared an Emergency Classification due to an Emergency Action Level (EAL) being exceeded.
· Degrading plant conditions may require Emergency Classification escalation.
· The Emergency Operations Facility (EOF) is staffed and ready to assume command and control.

In accordance with EP-AA-112, EMERGENCY RESPONSE ORGANIZATION/EMERGENCY RESPONSE FACILITY ACTIVATION AND OPERATION, which of the following responsibilities can be transferred to the EOF?

1. Event Classification
2. Protective Action Recommendations
3. Notification of Offsite Agencies (state and local)
4. Emergency Exposure Controls

A. 1 AND 3

B. 1 AND 4

C. 2 AND 3

D. 2 AND 4

Answer	C

Answer Explanation

A – Plausible: All four possible responses are specific duties of the emergency director. Event classification may be transferred from the Shift Emergency Director to the Station Emergency Director, but not to the EOF.

B – Plausible: All four possible responses are specific duties of the emergency director. Event classification and emergency exposure controls may be transferred from the Shift Emergency Director to the Station Emergency Director, but not to the EOF.

C – Correct: EP-AA-112, step 3.1, PARs and Notifications are transferred to EOF, Classification and Emergency Exposure Control remain at station.

D – All four possible responses are specific duties of the emergency director. Emergency exposure controls may be transferred from the Shift Emergency Director to the Station Emergency Director, but not to the EOF.

Question Information

Topic	Byron 2023 NRC Exam Q100				
User ID	SG40040-001			System ID	2498998
Status	Active	Point Value	1.00	Time (min)	0

Cross Reference Number	CLOSED
Cognitive Level	SRO-LOW
Operator Discipline	LO-I
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

NRC Exams Only

Question Type	Bank from Bwd 2020 NRC exam	**Difficulty**	N/A
Technical Reference and Revision #	EP-AA-112, Rev. 22		
Training Objective	7F.ZP-008-A		
Previous NRC Exams Use	None		

References Provided	None
K/A Justification	Meets the K/A by requiring examinee knowledge of SRO responsibilities in emergency plan implementation and which duties cannot be delegated to other facilities.
SRO-Only Justification	Question is SRO level because the Shift Emergency Director is the SM in the MCR which can only be performed by a licensed SRO.
Additional Information	None

K/A Reference(s)

P2.4.52	Safety Function	Tier 3	Group	RO Imp: 3.0	SRO Imp: 4.0
Knowledge of the lines of authority during implementation of the emergency plan, emergency plan implementing procedures, emergency operating procedures, or severe accident guidelines (CFR: 41.10 / 45.13)					

Learning Objective(s)

2023 NRC SRO Exam
User (Sys) ID N/A (1666691)

Cross Reference Links

None