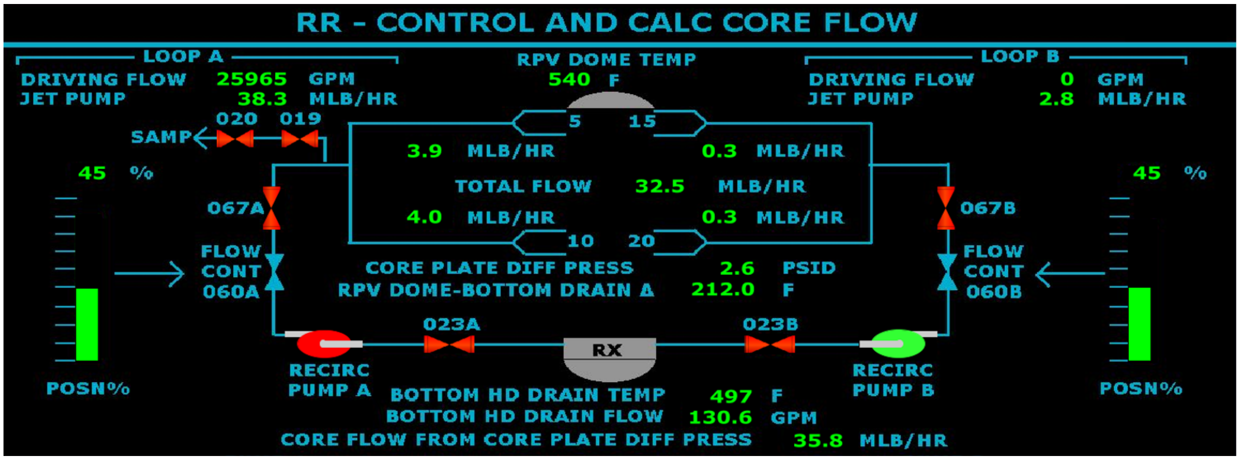
EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

1	ID: 2103926	Points: 1.00

The plant was operating at rated thermal power when the 'B' RR Pump tripped on overcurrent.

Reactor power is now stable at 55%.



What action is required?

A. Open RR FCV A to raise core flow.

B. Open RR FCV B to ~90% to prevent thermal stratification.

C. Place the mode switch in shutdown to exit the Restricted Zone.

D. Insert control rods in reverse rod sequence to exit the Controlled Entry Region.

Answer: D

Answer Justification / Plausibility Statements
D is correct. Per CPS 4008.01 Abnormal Reactor Coolant Flow, step 6.5.6.3 Core Flow Indication Guidance, when one RR Pump is running, the only valid indication of total core flow is via core plate dP due to reverse loop/jet pump flow inaccuracies. At 55% power and 35.8 mlb/hr (core flow from core plate diff press), the reactor is operating in the controlled entry region of the power/flow operating map. Per CPS 4008.01 step 6.5.4, an inadvertent or forced entry into the controlled entry region requires a prompt exit via reverse rod sequence or CRAM RODS. Incorrect Responses: A is incorrect but plausible. This action would result in exit from the Controlled Entry Region, but is prohibited by CPS 4008.01. B is incorrect but plausible. This action is required for an anticipated RR loop/pump recovery/restart to prevent thermal stratification prior to loop restart. With the RR Pump motor tripped on overcurrent, RR Pump 'B' recovery/restart is not anticipated and the action is therefore not required. C is incorrect but plausible. The intersection of 32.5 mlb/hr (total flow) and 55% power is in the restricted zone of the power to flow map. Operation in the restricted zone of the power to flow map is prohibited and requires a scram. The total flow value is not accurate in single loop operations.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295001.AA2.03	AA2.03	3.3	3.3	1		1

System Name
Partial or Complete Loss of Forced Core Flow Circulation

Category Statement
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
Actual core flow

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q1 295001 A2.03

Other NRC Data

References Provided	CPS 3005.01 Unit Power Changes Rev. 43f Figure 1: Stability Control & Power/Flow Operating Map		
K/A Justification	This question meets the KA because the examinee has to interpret actual core flow in the stem graphic to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and application level. The examinee must analyze the parameters in a graphic and then determine required actions based on the analysis (3-SPK/SPR).		
NRC Exams Only			
Question Type	Bank (CL-LC-1838)	**Difficulty**	N/A
Technical Reference and Revision #	CPS 4008.01 Rev. 20d		
Training Objective	DB400801.01.04 Given CPS No. 4008.01, ABNORMAL REACTOR COOLANT FLOW, describe the methods to be used to exit Controlled Entry Region.		
Previous NRC Exam Use	None		

2	ID: 896308	Points: 1.00

Plant conditions are as follows:

- Plant is in Mode 5 with the Main Steam Line Plugs installed.
- 4160V Bus 1A1 is deenergized for scheduled outage activities.
- RHR 'B' is operating in Shutdown Cooling Mode.

THEN, 4160V Bus 1B1 deenergized due to a fault.

Which of the following shutdown cooling methods listed in CPS 4006.01 Loss of Shutdown Cooling Table 2: Shutdown Cooling Methods is/are immediately available for decay heat removal (can be aligned for decay heat removal from the MCR)?

Method 1	ECCS Feed and Bleed Through SRVs
Method 2	Reactor Water Cleanup
Method 3	Fuel Pool Cooling and Cleanup Using Natural Circulation

A.	Method 1 ONLY
B.	Method 2 ONLY
C.	Methods 1 and 2 ONLY
D.	Methods 2 and 3 ONLY

Answer:	B

Answer Justification / Plausibility Statements
B is correct: With a loss of 4160V Bus 1A1 and 1B1, the only methods listed that are not affected by the loss of power are Methods 1 and 2. • ECCS Feed and Bleed Through SRVs can be accomplished by operating the HPCS Pump (which is powered from 4160V Bus 1C1) and flowing water through the SRVs. • RWCU can be aligned for alternate shutdown cooling IAW CPS 3303.01 Reactor Water Cleanup (RT) section 8.2.6 by bypassing the Regenerative Heat Exchangers and using the NRHXs for a heat sink. CCW and RT must be in operation; neither system requires power from 4160V Bus 1A1 or 1B1 to support alternate decay heat removal operations. With the Main Steam Line Plugs installed, the flowpath through the SRVs cannot be established without removal, and therefore cannot be aligned from the MCR. This leaves Method 2 (Reactor Water Cleanup) as the only remaining decay heat removal method. Incorrect Responses: A is incorrect but plausible because ECCS Feed and Bleed Through SRVs is partially available. High Pressure Core Spray can be used to provide flow (is unaffected by loss of power), but the SRVs are not available with the Main Steam Line Plugs installed. C is incorrect but plausible because methods 1 and 2 are listed as alternate decay heat removal systems in CPS 4006.01 Table 2: Shutdown Cooling Methods. Method 1, however, cannot be aligned from the MCR with the Main Steam Line Plugs installed. D is incorrect but plausible and is partially correct. Method 2 (Reactor Water Cleanup) is immediately available as an alternate decay heat removal system. Method 3 (FC Using Natural Circulation) is NOT immediately available. Although natural circulation flow does not normally require AC power, CPS 3312.02 step 8.1.5 states that both FC Pumps have to be running to establish cooling. With 4160V Bus 1A1 and 1B1 deenergized, the FC Pumps will have no power available, so Method 3 cannot be used.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295003	B2.4.09	3.8	4.2	1	1	N/A

System Name
Partial or Complete Loss of A.C. Power

Category Statement
Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q2 295003 2.4.9

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee is required to determine mitigating strategies to be used for a loss of shutdown cooling during a partial loss of AC power event to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the conditions in the stem and then determine which systems are available for alternate shutdown cooling based on the analysis (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-N12017)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3303.01 Rev. 37b • CPS 4006.01 Rev. 5c • CPS 3312.02 Rev. 9d		
Training Objective	DB400601.01.04 Recall the different methods to remove decay heat from the reactor core.		
Previous NRC Exam Use	ILT 12-1 NRC		

3	ID: 2104503	Points: 1.00

The plant has experienced a station blackout.

Load shedding was performed within the time required by CPS 4200.01 Loss of AC Power.

Div 1 DC power will be maintained to _____(1)_____ for a minimum of _____(2)_____ hours.

A.	(1) Reactor Core Isolation Cooling ONLY (2) 4
B.	(1) Reactor Core Isolation Cooling ONLY (2) 8
C.	(1) Reactor Core Isolation Cooling AND Safety Relief Valves (2) 4
D.	(1) Reactor Core Isolation Cooling AND Safety Relief Valves (2) 8

Answer:	C

Answer Justification / Plausibility Statements
C is correct: Per CPS 4200.01 Loss of AC Power, step 6.2.4 - the completion of DC load shedding within 1 hour is a TIME CRITICAL ACTION. Essential control circuits are powered from station batteries. Battery loads are reduced to insure a minimum 4 hour DC supply to maintain manual SRV control and RCIC operation. Incorrect Responses: A is incorrect but plausible. This answer is partially correct because DC power will be maintained to RCIC during the coping period. The second part is correct. B is incorrect but plausible: • Part 1 is partially correct because DC power will be maintained to RCIC during the coping period. • Part 2 is also partially correct because load shedding will lengthen the coping period. Per CC-CL-118 SITE IMPLEMENTATION OF DIVERSE AND FLEXIBLE COPING STRATEGIES (FLEX) AND SPENT FUEL, 8 hours is the length of time the "Safe Shutdown" emergency light batteries are expected to last. D is incorrect but plausible because load shedding will lengthen the coping period. Per CC-CL-118 SITE IMPLEMENTATION OF DIVERSE AND FLEXIBLE COPING STRATEGIES (FLEX) AND SPENT FUEL, 8 hours is the length of time the "Safe Shutdown" emergency light batteries are expected to last. The first part is correct.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295004.AK1.04	AK1.04	2.8	2.9	1		6

System Name
Partial or Complete Loss of D.C. Power

Category Statement
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : (CFR: 41.8 to 41.10)

K/A Statement
Effect of battery discharge rate on capacity

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10
41.8	41.8
41.9	41.9

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

263000.01	263000.01 STATE the purpose(s) of the BATTERY & DC DISTRIBUTION System including applicable design bases.

263000.01	263000.01 STATE the purpose(s) of the BATTERY & DC DISTRIBUTION System including applicable design bases.

	Q3 295004 K1.04

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must display knowledge of the operational implications of the actions taken to minimize battery discharge rate during a partial loss of DC power to answer this question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall facts from a procedure to answer this question (1-F).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A12005)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4200.01 Rev. 26 • CC-CL-118 Rev. 1c		
Training Objective	263000.01 STATE the purpose(s) of the BATTERY & DC DISTRIBUTION System including applicable design bases.		
Previous NRC Exam Use	None		

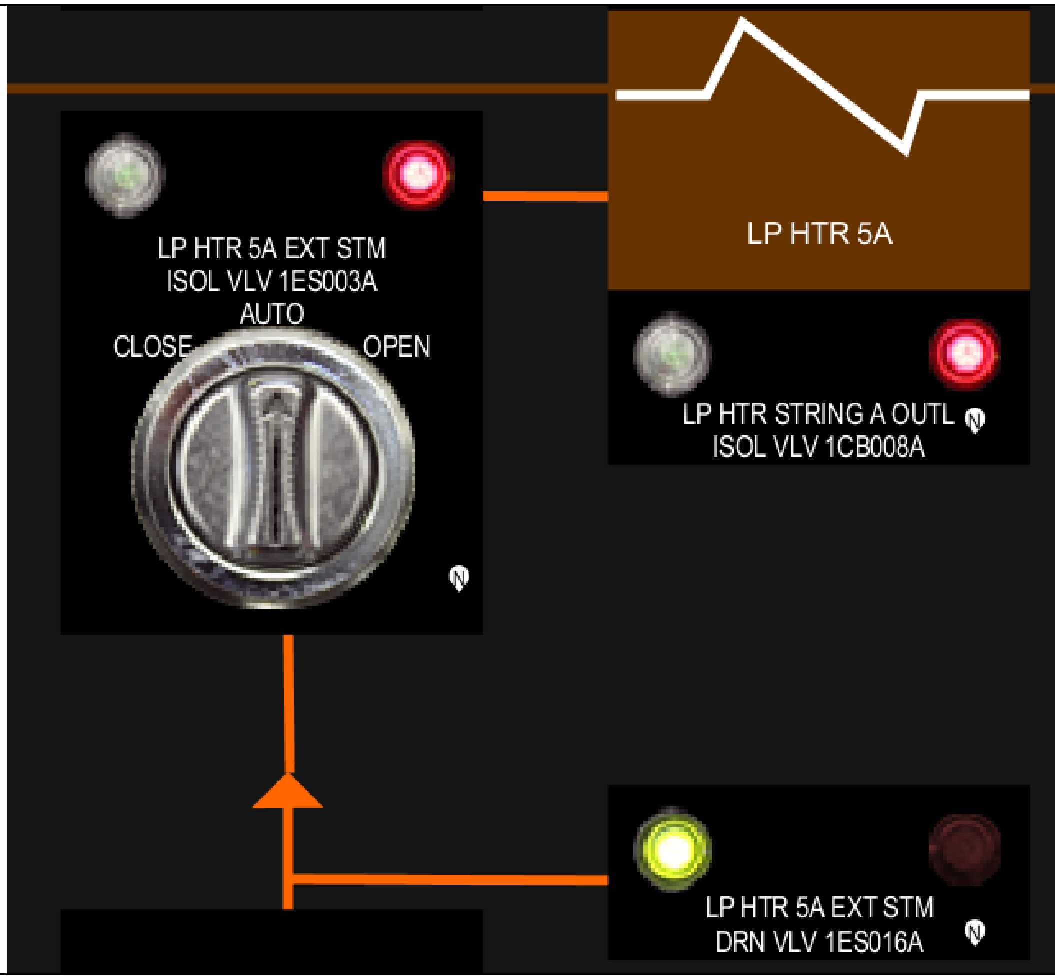
4	ID: 2104508	Points: 1.00

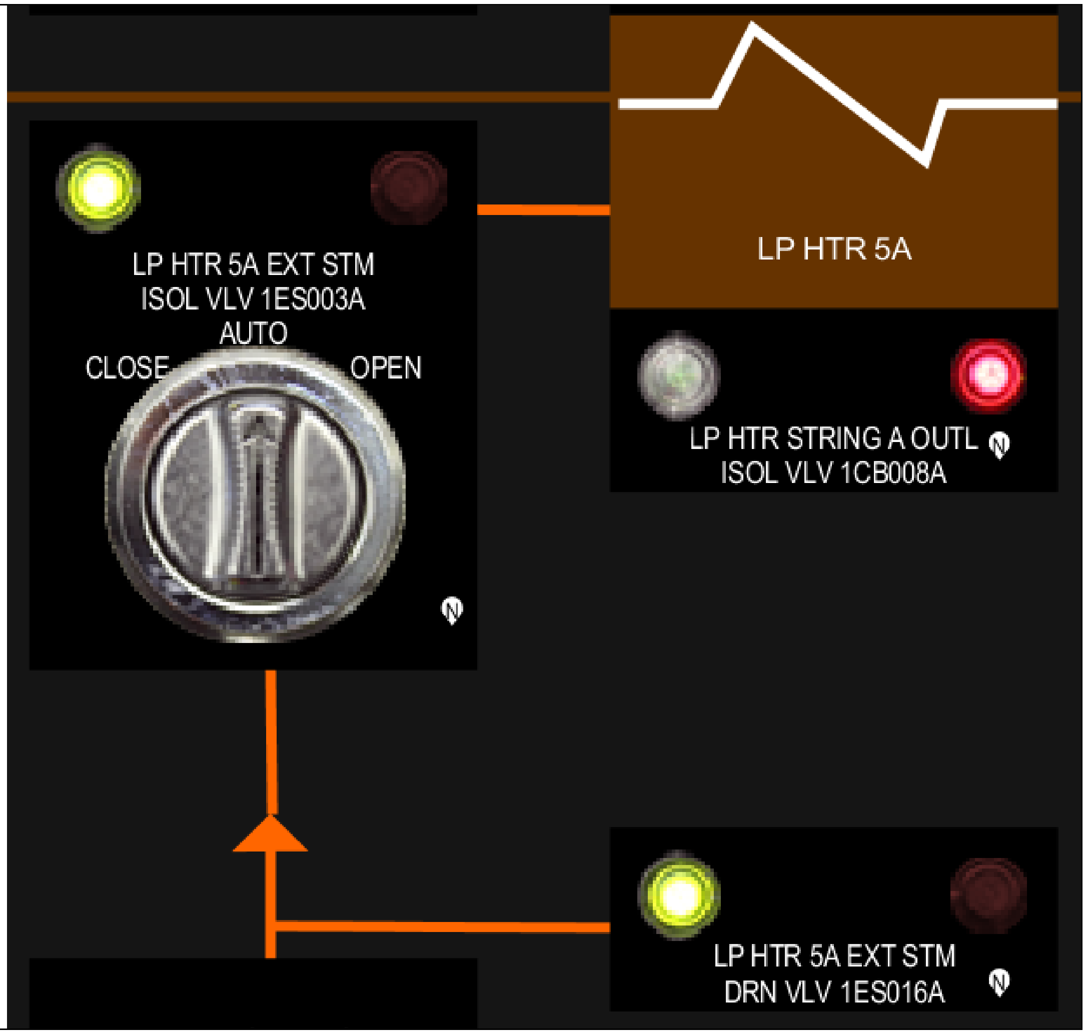
The plant was operating at rated thermal power.

THEN, the Main Turbine tripped.

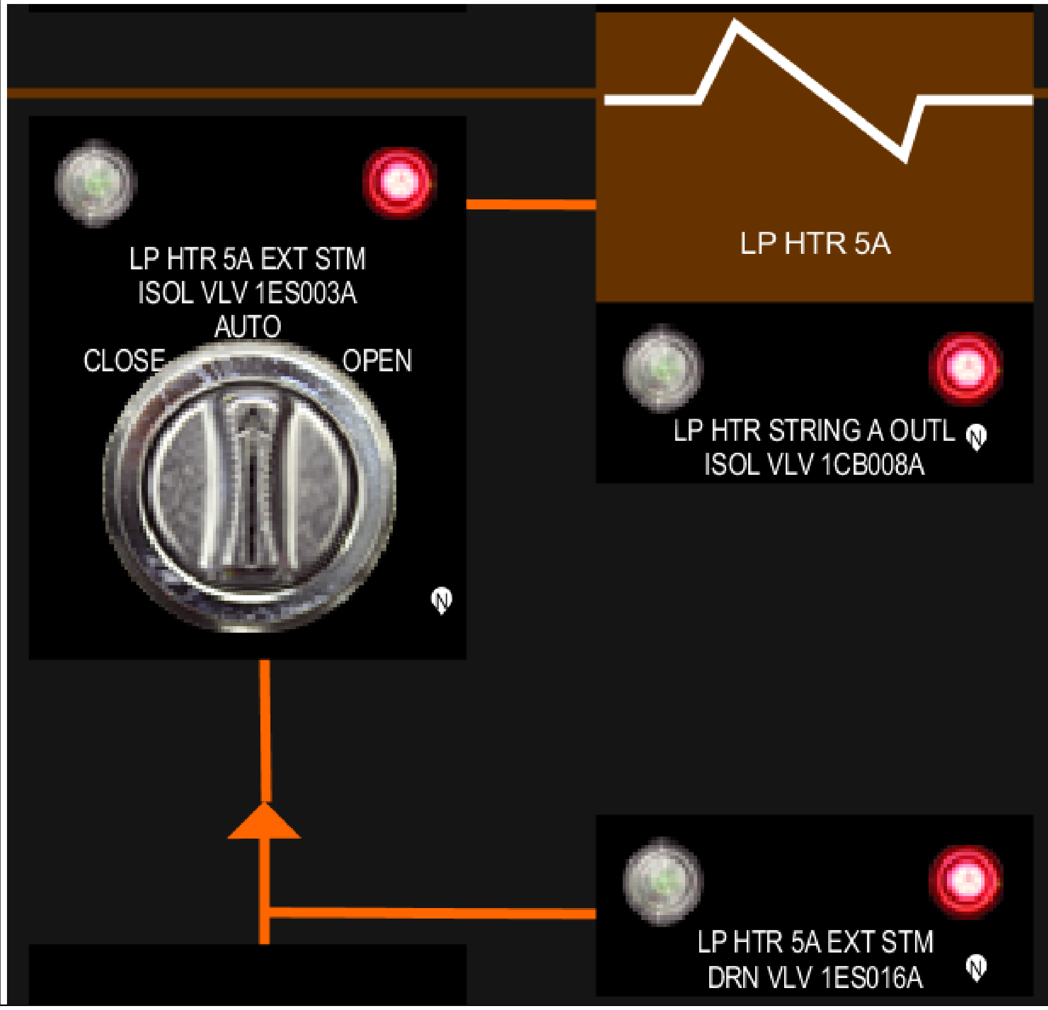
Which of the following shows the expected positions for 1ES003A LP HTR 5A EXT STM ISOL VLV and 1ES016A LP HTR 5A EXT STM DRN VLV one minute later?

Alignment 1

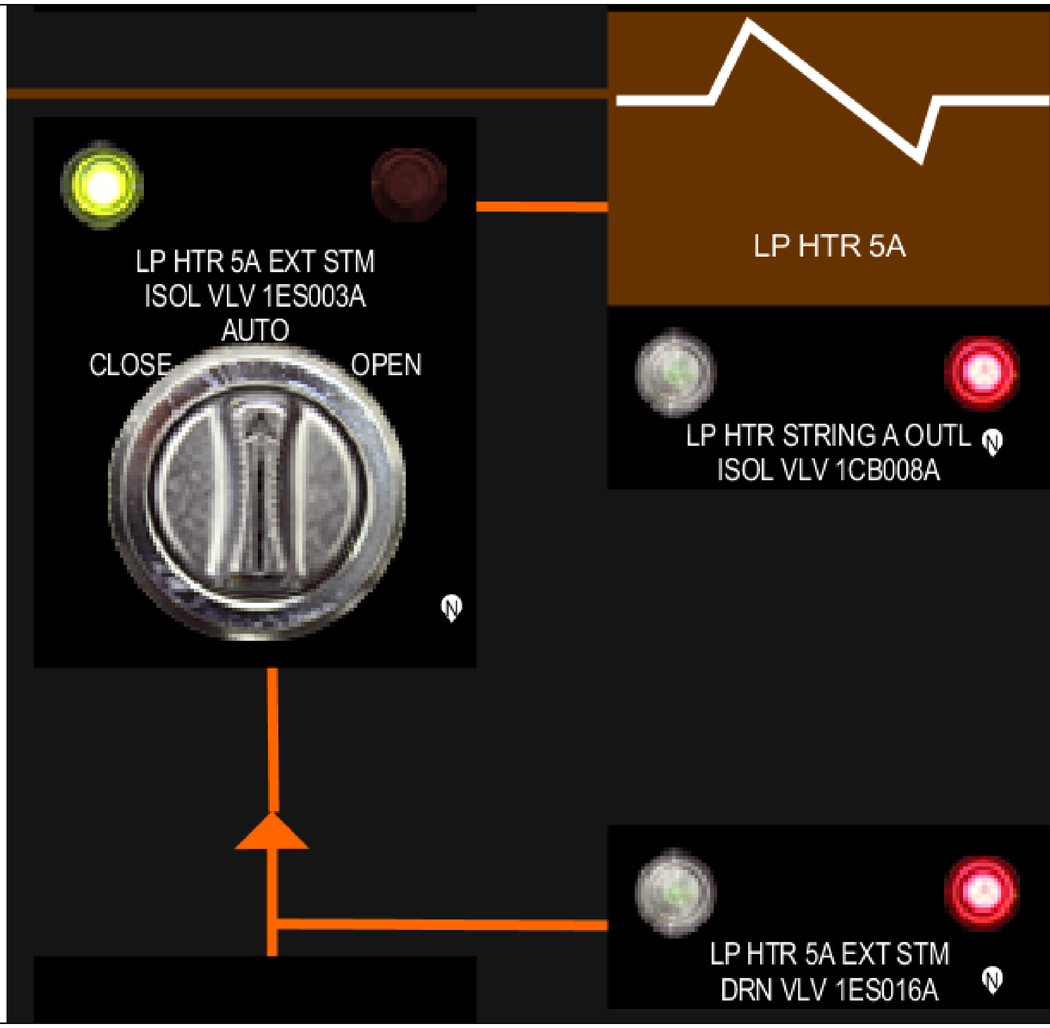


Alignment 2	

LP HTR 5A EXT STM
ISOL VLV 1ES003A
AUTO
CLOSE
OPEN
LP HTR 5A
LP HTR STRING A OUTL
ISOL VLV 1CB008A
LP HTR 5A EXT STM
DRN VLV 1ES016A

Alignment 3	

LP HTR 5A EXT STM
ISOL VLV 1ES003A
AUTO
CLOSE
OPEN
LP HTR 5A
LP HTR STRING A OUTL
ISOL VLV 1CB008A
LP HTR 5A EXT STM
DRN VLV 1ES016A

Alignment 4	

A.	Alignment 1
B.	Alignment 2
C.	Alignment 3
D.	Alignment 4

Answer:	D

Answer Justification / Plausibility Statements
D is correct. Per 5015-4L Closed LP HTR 5A ES CHECK VLV, a Main Turbine trip will cause 1ES003A to automatically close and 1ES016A to automatically open. This is depicted in Alignment 4. Incorrect Responses: A is incorrect but plausible because Alignment 1 is the normal alignment for 1ES003A and 1ES016A without a Turbine Trip. B is incorrect but plausible and would be correct if all ES valves (including 1ES003A and 1ES016A) behaved like automatic containment isolation valves which automatically close on an isolation signal. C is incorrect but plausible and would be correct if all ES valves (including 1ES003A and 1ES016A) behaved like the MSL Drain Valves which automatically open during low steam flow conditions to keep the MSLs drained.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295005.AK2.05	AK2.05	2.6	2.7	1		3

System Name
Main Turbine Generator Trip

Category Statement
Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: (CFR: 41.7 / 45.8)

K/A Statement
Extraction steam system

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q4 295005 K2.05

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of how a turbine trip impacts the extraction steam isolation and drain valves for a low pressure feedwater heater to answer the question.
SRO-Only Justification	N/A
Additional Information	Cog Level Justification - this is a low cog question written at the memory level. The examinee has to recall the interlocks for the 5A FW Heater ES valves to answer the question (1-I).
NRC Exams Only	
Question Type	New — **Difficulty** N/A
Technical Reference and Revision #	CPS 5015.04 (4L) Rev. 24
Training Objective	239003.03 DESCRIBE the function, operation, interlocks, trips, and power supplies of the following EXTRACTION STEAM, HEATER VENTS & DRAINS System components. .4 Extraction Steam Isolation Valves
Previous NRC Exam Use	None

5	ID: 2107214	Points: 1.00

The plant was operating at rated thermal power.

THEN, a reactor scram occurred due to a turbine trip.

Which of the following cases describes the:

1) Reactor Recirculation (RR) Pump response to the Turbine Trip / Reactor Scram, <u>and</u>

2) the reason for the RR Pump response?

	RR Pump Response	Reason
Case 1	Immediately downshifts to slow speed.	Protects the RR Pumps from cavitation.
Case 2	Immediately downshifts to slow speed.	Ensures MCPR Safety Limit is not exceeded at the end-of-cycle.
Case 3	Downshifts to slow speed when RPV level reaches Level 3.	Protects the RR Pumps from cavitation.
Case 4	Downshifts to slow speed when RPV level reaches Level 3.	Ensures MCPR Safety Limit is not exceeded at the end-of-cycle.

A.	Case 1
B.	Case 2
C.	Case 3
D.	Case 4

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per ITS B3.3.1.1 RPS Instrumentation (B3.3-16), the bases for Turbine Stop Valve Closure RPS trip is stated as follows: Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 4. For this event, the reactor scram reduces the amount of energy required to be absorbed and, **along with the actions of the End of Cycle Recirculation Pump Trip (EOC RPT) System, ensures that the MCPR SL is not exceeded.** Per CPS 5003.02 (2F) RECIRC MTR A AUTO TRIP OR XFR TO LS, a trip or load reject of main turbine when ≥ 33.3% power will initiate a trip of 5A (fast speed) breaker, and an auto transfer to Low Frequency Motor Generator (slow speed). Incorrect Responses: A is incorrect but plausible because the RR Pumps will downshift to slow speed on a TSV closure and at RPV Level 3. Part 2 is incorrect because the EOC-RPT signal occurs first. By the time RPV level lowers to Level 3, the RR Pumps will be in slow speed and cavitation is no longer a concern. C is incorrect but plausible because the RR Pumps will downshift to slow at Level 3 to protect the RR Pumps from cavitation. The downshift occurs first due to the Main Turbine trip / Reactor scram. The downshift adds negative reactivity to the core to ensure the MCPR SL is not exceeded at the end-of-cycle. D is incorrect but plausible because the RR Pumps will downshift to slow at Level 3 to protect the RR Pumps from cavitation. The downshift occurs first due to the Main Turbine trip / Reactor scram.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295006.AK3.06	AK3.06	3.2	3.3	1		1

System Name
SCRAM

Category Statement
Knowledge of the reasons for the following responses as they apply to SCRAM : (CFR: 41.5 / 45.6)

K/A Statement
Recirculation pump speed reduction: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q5 295006 K3.06

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to demonstrate knowledge of the reason for EOC-RPT actuation during a scram event to answer the question.		
SRO Only Justification Statement:	N/A		
Additional Information:	Question is written at the memory level and requires recall of system facts (1-F).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 5003.02 (2F) Rev. 35c • ITS B3.3.1.1 (B3.3-16) Rev. 7-5		
Training Objective:	202001.10 EXPLAIN the reasons for given Reactor Recirculation System operating limits and precautions. .7 End of Cycle (EOC) Pump trip		
Previous NRC Exam Use:	None		

6	ID: 2104509	Points: 1.00

Which of the following cases describe the impact to Diesel Generator 1A when placing Transfer Switch C61-HS502 on the Remote Shutdown Panel in the EMERG position?

	Will automatically start on high DW pressure?	**Will automatically start on its associated 4KV bus undervoltage?**
Case 1	N	N
Case 2	Y	Y
Case 3	N	Y
Case 4	Y	N

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

Answer:	C

Answer Justification / Plausibility Statements
C is correct: Per CPS 4003.01C004 RSP - Diesel Generator 1A Operation, section 3.0 Limitations states: When TRANSFER SWITCH C61-HS502 is in EMERG: • DG 1A is prevented from starting on a LOCA signal. • DG 1A normal safety trips are bypassed with the exception of Overspeed & Gen Differential 87 relay. • DG 1A will only start on 4160V Bux 1A1 under voltage. This is described in Case 3. Incorrect Responses: A is incorrect but plausible. Part 1 is correct. Part 2 would be correct if the 4KV Bus UV start was overridden like the LOCA start signal with C61-HS502 in EMERG. B is incorrect but plausible. Part 2 is correct (DG 1A will automatically start on 1A1 undervoltage). Part 1 would be correct if LOCA start signals behaved like UV start signals with C61-HS502 in EMERG. D is incorrect but plausible if the 4KV Bus UV start signal behaved like the LOCA start signal, and vice versa.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295016.AA1.04	AA1.04	3.1	3.2	1		7

System Name
Control Room Abandonment

Category Statement
Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT : (CFR: 41.7 / 45.6)

K/A Statement
A.C. electrical distribution

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

200000.03	200000.03 DESCRIBE the function, operation, interlocks, trips, and power supplies of the following REMOTE SHUTDOWN System components. .1 Residual Heat Removal (RHR) Pump Minimum Flow Control Valve Logic .2 Safety Relief Valves (SRVs) Relief Mode when Transfer Switch 1C61-S10 is in the Emergency position .3 Remote Shutdown Panel Transfer Switches

	Q6 295016 A1.04

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must demonstrate the ability to monitor the AC electrical distribution by anticipating the effect of placing the Div 1 DG RSP transfer switch in emergency during Control Room Abandonment to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall singular system response to a switch placement to answer the question (1-I).		
NRC Exams Only			
Question Type	Bank (CL-ILT-0129)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4003.01C004 Rev. 1e • CPS 3506.01 Rev. 39c		
Training Objective	200000.03 DESCRIBE the function, operation, interlocks, trips, and power supplies of the following REMOTE SHUTDOWN System components. .3 Remote Shutdown Panel Transfer Switches		
Previous NRC Exam Use	None		

7	ID: 2104511	Points: 1.00

The plant was operating at rated thermal power.

Component Cooling Water (CCW) Pump 'A' is out of service.

THEN, a loss of power transient occurred resulting in the following steady state indication.

Which of the following actions are required?

A.	Scram the plant and perform a manual Group 1 isolation.
B.	Shutdown Reactor Recirculation (RR) pumps within one minute.
C.	Shutdown and isolate the Reactor Water Cleanup (RT) system within one minute.
D.	Shutdown the running Fuel Pool Cooling (FC) pump and isolate the FC Heat Exchangers.

Answer:	D

Answer Justification / Plausibility Statements
D is correct: The normal alignment for the CCW system is 2 pumps running with CCW pressure in the green band (~ 90 psig). The steady state pressure indication provided in the stem indicates that a partial loss of CCW has occurred. IAW CPS 5040.01 (1B), if only one CCW Pump is running, then start a standby CCW Pump , <u>or</u> shutdown the running FC Pump and shut 1CC076A & 76B, FC Heat Exchanger Outlet Valves. Since the stem states that the third CCW Pump is out of service, the only remaining available action is to shutdown the running FC Pump and isolate the FC Heat Exchangers. Incorrect Responses: A is incorrect but plausible. Per CPS 3203.01 section 8.3.6, a loss of CCW will cause an eventual loss of the Service Air Compressors which supply instrument air to the Main Steam Isolation Valves (MSIVs). The MSIVs will then fail closed on a loss of IA. B is incorrect but plausible. Per 5040-1B, a <u>complete</u> loss of CCW (<u>no</u> CCW Pumps running) requires the RR Pumps to be secured within 1 minute (protects the RR Pump Motors and seals). C is incorrect but plausible. Per CPS 3203.01 Appendix B CCW Loads, CCW supplies the Reactor Water Cleanup Pumps. Per CPS 3303.01 section 8.3.3 CAUTION, RWCU cannot be operated with a complete loss of Component Cooling Water to prevent damage to the RWCU Filter Demins and pumps seals, but the pumps are not required to be secured within one minute on a partial loss of CCW. The 1 minute time requirement pertains to a complete loss of CCW to the RR Pumps. In addition, annunciator 5000-2E Cleanup Pump Seal Gland Plate Temp Hi requires RWCU Pumps to be removed from service if 5000-2E is received and temperature cannot be lowered (no time requirement specified).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295018.AA2.05	AA2.05	2.9	2.9	1		8

System Name
Partial or Complete Loss of Component Cooling Water

Category Statement
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
System pressure

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q7 295018 A2.05

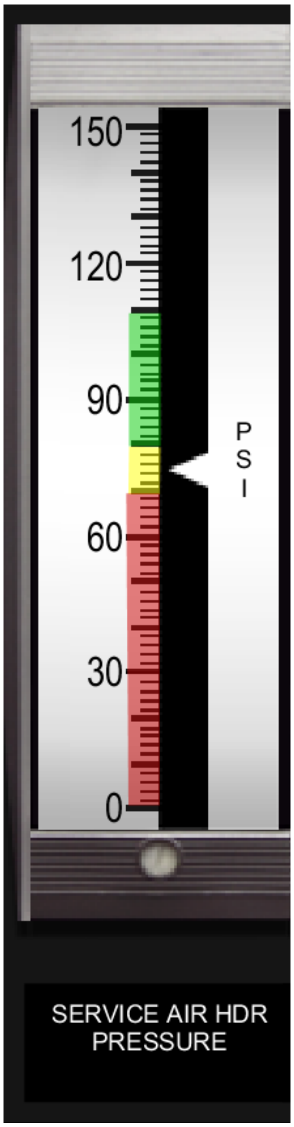
Other NRC Data

References Provided	None		
K/A Justification	KA Justification - this question meets the KA because the candidate must interpret the CCW system pressure indication provided in the stem and then determine required actions to answer the question.		
SRO-Only Justification	N/A		
Additional Information	Cog Level Justification - this is a high cog question written at the analysis and comprehension level. The examinee has to interpret the indications provided in the stem of the question and then determine the appropriate actions based on that analysis.		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5040.01 (1B) Rev. 28c • CPS 3203.01 Rev. 35e • CPS 3303.01 Rev. 37b • CPS 5000.02 (2E) Rev. 25b		
Training Objective	400001.09 DISCUSS the effect: A total loss or malfunction of the Component Cooling Water System has on the plant. A total loss or malfunction of various plant systems has on the Component Cooling Water System. .1 Loss of Component Cooling Water on CC loads (including RR pump seals)		
Previous NRC Exam Use	None		

8	ID: 2104645	Points: 1.00

The plant was operating at rated thermal power.

THEN, a transient occurred resulting in the following indication.



Assume the indicated value is the LOWEST value reached and there has been NO operator action.

Which of the following cases show the annunciator status for current plant conditions?

Case 1	**LOW PRESS RADWASTE BLDG SERV AIR HDR** OFF **AUTO START SERVICE AIR COMPRESSOR** OFF
Case 2	**LOW PRESS RADWASTE BLDG SERV AIR HDR** ON **AUTO START SERVICE AIR COMPRESSOR** OFF
Case 3	**LOW PRESS RADWASTE BLDG SERV AIR HDR** OFF **AUTO START SERVICE AIR COMPRESSOR** ON

Case 4	**LOW PRESS** **RADWASTE BLDG** **SERV AIR HDR** ON **AUTO START** **SERVICE AIR** **COMPRESSOR** ON

A.	Case 1
B.	Case 2
C.	Case 3
D.	Case 4

Answer:	C

Answer Justification / Plausibility Statements
C is correct: The graphic in the stem indicates that Service Air Header pressure has degraded to ~ 75 psig. Per CPS 5041.06 (6B) Auto Start Service Air Compressor, the annunciator is actuated if service air header pressure lowers to 80 psig. Incorrect Responses A is incorrect but plausible. Case 1 indicates that neither of the annunciators in the graphic are in alarm which would be correct if Service Air Header Pressure was above 80 psig. Also plausible with Service Air Header Pressure indicating in the yellow band of the meter. B is incorrect but plausible. This answer would be correct if building low service air header pressure alarms preceded the auto start of the standby Service Air Compressor. Incorrect because the setpoint for 5041-5B is 70 psig. Air header pressure has not yet degraded to this value. D is incorrect but plausible. This answer would be correct if the various building low pressure alarms and the service air compressor auto started at the same setpoint.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295019	B2.4.46	4.2	4.2	1	1	N/A

System Name
Partial or Complete Loss of Instrument Air

Category Statement
Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q8 295019 2.4.46

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to verify that the alarms are consistent with the plant conditions (partial loss of instrument air) to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the indication provided in the stem and then determine which annunciators should be in alarm to answer the question (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5041.05 (5B) Rev. 29a • CPS 5041.06 (6B) Rev. 31		
Training Objective	300000.03 DESCRIBE the function, operation, interlocks, trips, physical locations and power supplies of the following Service and Instrument Air System components. .1 Service Air Compressors .8 Ring Headers		
Previous NRC Exam Use	None		

9	ID: 2104648	Points: 1.00

The Reactor was shutdown TEN days ago.

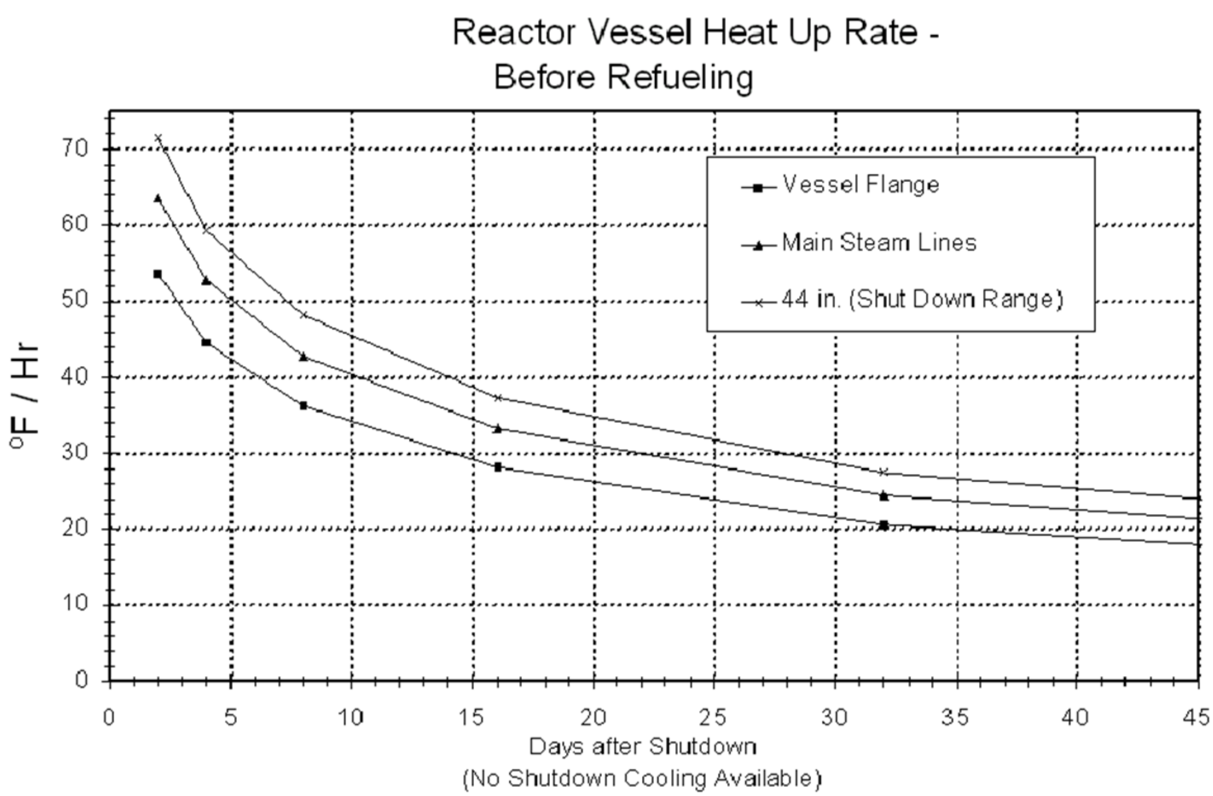
The plant entered Mode 5 THREE days ago.

Current plant conditions:

- RHR Loop 'A' is in Shutdown Cooling.
- RPV level is at the natural circulation level.
- Coolant temperature is 110°F.

THEN, a loss of shutdown cooling occurred.

With no operator actions, Mode 3 temperature conditions will be reached in approximately ______ hours.



A.	1.3 - 1.5
B.	1.6 - 1.8
C.	1.9 - 2.1
D.	2.5 - 2.7

Answer:	C

Answer Justification / Plausibility Statements
C is correct: Per CPS 3312.03 RHR - Shutdown Cooling (SDC) & Fuel Pool Cooling and Assist (FPC&A), step 4.2 states that the natural circulation level is 44" - Shutdown Range or 61" - Upset Range during Modes 4/5. 44" Shutdown Range is represented by the top line on the heatup rate curve provided in the stem. The intersection of the top curve and 10 days after shutdown yields a heatup rate of ~ 46°F/hr. Per ITS Table 1.1-1, Mode 3 is defined as hot shutdown with coolant temperature > 200°F. With initial coolant temperature at 110°F, Mode 3 will be reached in 1.96 hours. (200 - 110) / 46 = 1.96 hours. Incorrect Responses: A is incorrect but plausible. A heatup rate of 65°F/hr (intersection of 3 days and the top curve) would result in Mode 3 being reached in 1.38 hours. (200 - 110) / 65 = 1.38 hours. B is incorrect but plausible. A heatup rate of 50°F/hr (intersection of 3 days and the bottom curve) would result in Mode 3 being reached in 1.8 hours. (200 - 110) / 50 = 1.8 hours. D is incorrect but plausible. A heatup rate of 35°F/hr (intersection of 10 days and the bottom curve) would result in Mode 3 being reached in 2.57 hours. (200 - 110) / 35 = 2.57 hours.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295021.AK1.04	AK1.04	3.6	3.7	1		4

System Name
Loss of Shutdown Cooling

Category Statement
Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : (CFR: 41.8 to 41.10)

K/A Statement
Natural circulation

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10
41.8	41.8
41.9	41.9

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q9 295021 K1.04

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to determine the time it takes to reach Mode 3 (the operational implication) in the event of a loss of shutdown cooling) with RPV level at the natural circulation level to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to calculate the RPV heatup rate based on conditions provided in the stem to answer the question (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3312.03 Rev. 11e • ITS 1.0 (Table 1.1-1) Amendment No. 95		
Training Objective	DB400601 Loss of Shutdown Cooling (4006.01) Identify the following conditions: .2 Mode 3 entry		
Previous NRC Exam Use	None		

10	ID: 2107064	Points: 1.00

Refueling operations were in progress when a fuel bundle was dropped in the Spent Fuel Pool.

THEN, a red indicator tile and audible alarm was observed on the MCR AR/PR LAN for 1RIX-PR019 Fuel Building CAM.

The Accident Range HVAC Monitor, 0RIX-PR012 will be in ___(1)___.

Entry into EOP-8 Secondary Containment Control _____(2)_____ required.

A.	(1) standby (2) is

B.	(1) standby (2) is NOT

C.	(1) operation (2) is

D.	(1) operation (2) is NOT

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Part 1 - 0RIX-PR012 will only auto start on a high alarm on the noble gas channel for 0RIX-PR001 or 2 HVAC Exhaust PRM. A high alarm on 1RIX-PR019 has no effect on the 0RIX-PR012. Part 2 - Per CPS 4406.01 EOP-8 Secondary Containment Control, a high alarm on 1RIX-PR019 is an entry condition for EOP-8. Incorrect Responses: B is incorrect but plausible. The first part is correct. Many of the PRM entries for EOP-8 (area radiation above max normal) are determined by local surveys, not by CAM alarms. C is incorrect but plausible. 0RIX-PR012 is automatically started when 0RIX-PR001 or 2 Channel 1, 3, or 5 reaches a high alarm setpoint. There are no automatic functions for 0RIX-PR012. The second part is correct. D is incorrect but plausible: • 0RIX-PR012 is automatically started when 0RIX-PR001 or 2 Channel 1, 3, or 5 reaches a high alarm setpoint. There are no automatic functions for 0RIX-PR012, and • Many of the PRM entries for EOP-8 (area radiation above max normal) are determined by local surveys, not by CAM alarms.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295023.AK2.03	AK2.03	3.4	3.6	1		8

System Name
Refueling Accidents

Category Statement
Knowledge of the interrelations between REFUELING ACCIDENTS and the following: (CFR: 41.7 / 45.8)

K/A Statement
Radiation monitoring equipment

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q10 295023 K2.03

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of the interrelationship between refueling accidents and FB Cam 1RIX-PR019 to answer the question.
SRO-Only Justification	N/A
Additional Information	This is a low cog question written at the memory level. The examinee has to recall an EOP entry condition and system interlocks to answer the question (1-F/1-I).
NRC Exams Only	
Question Type	New — **Difficulty** N/A
Technical Reference and Revision #	• CPS 5140.41 Rev. 1G • CPS 5140.31 Rev. 0b • CPS 4406.01 Rev. 30
Training Objective	272000.06 Given an AR/PR System annunciator, DESCRIBE: a. The condition causing the annunciator b. Any automatic actions c. Any operational implications
Previous NRC Exam Use	None

11	ID: 2104660	Points: 1.00

A Loss of Coolant Accident is in progress.

Plant conditions are as follows:

- Containment pressure at 3.2 psig, trending up.
- Containment temperature at 150°F, trending up.
- Drywell temperature at 315°F, trending up.

Which of the following states the bases for performing a blowdown under these circumstances?

A blowdown is performed...

A.	to prevent exceeding the drywell design temperature limit.
B.	to prevent exceeding the containment design temperature limit.
C.	to ensure continued operability of RPV water level instrumentation.
D.	because the pressure suppression function of the containment is being challenged.

Answer:	D

Answer Justification / Plausibility Statements

D is correct:

A blowdown is performed per EOP-6 if Containment pressure cannot be maintained within Fig. N, Pressure Suppression Pressure. At 20' in the suppression pool, the Fig. N limit for containment pressure is 3.2 psig.

Per the EOP-TB discussion on EOP-6, the blowdown is performed to limit the release of energy into the containment, thus minimizing further increases in containment pressure.

The "Pressure Suppression Pressure" is imposed to ensure that the pressure suppression function of the containment is maintained while the RPV is at pressure. The Pressure Suppression Pressure is a function of suppression pool level and is defined to be the lesser of:

- The highest containment pressure which can occur without steam in the containment (limiting for CPS).
- The highest containment pressure from which a blowdown will not exceed the Primary Containment Pressure Limit before RPV pressure drops to the Decay Heat Removal Pressure.
- The highest containment pressure at which SRVs can be opened without exceeding the suppression pool boundary design load.

Incorrect Responses:

A is incorrect but plausible with drywell temperature rising, but the drywell temperature limit requiring a blowdown to be performed is 340°F and has not yet been reached.

B is incorrect but plausible with containment temperature rising, but the containment temperature limit requiring a blowdown to be performed is 185°F and has not yet been reached.

C is incorrect but plausible. RPV level indications are affected by high DW and Cnmt temperatures, but RPV flooding is not performed until RPV water level is unknown. This will not occur with Cnmt temperature at 150°F.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295024.EK3.04	EK3.04	3.7	4.1	1		5

System Name
High Drywell Pressure

Category Statement
Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : (CFR: 41.5 / 45.6)

K/A Statement
†Emergency depressurization

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q11 295024 K3.04

LP87558.01.02	LP87558.01.02 Given a diagram of EOP-6: .01 State the conditions to exit/transfer from EOP-6 .02 State the bases for each individual step/action of EOP-6 .03 State the systems that are available for Containment Purge .04 State the systems and flow paths that can be used to cool and control pressure in the containment.

LP87558.01.02	N-CL-OPS-DB-LP87558.01.02 Given a diagram of EOP-6: .01 State the conditions to exit/transfer from EOP-6 .02 State the bases for each individual step/action of EOP-6 .03 State the systems that are available for Containment Purge .04 State the systems and flow paths that can be used to cool and control pressure in the containment.

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee is required to know why a blowdown is performed when High Drywell Pressure conditions have caused containment pressure to exceed PSP.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the conditions in the stem and then determine the reason for required actions to answer the question (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-N12008)	**Difficulty**	N/A
Technical Reference and Revision #	EOP-TB Rev. 7		
Training Objective	N-CL-OPS-DB-LP87558.01.02 Given a diagram of EOP-6: .01 State the conditions to exit/transfer from EOP-6 .02 State the bases for each individual step/action of EOP-6 .03 State the systems that are available for Containment Purge .04 State the systems and flow paths that can be used to cool and control pressure in the containment.		
Previous NRC Exam Use	ILT 12-1 NRC Exam		

12	ID: 2104664	Points: 1.00

The plant was operating at rated thermal power.

THEN, a LOCA occurred requiring entry into EOP-1 RPV Control and EOP-3 Emergency RPV Depressurization (Blowdown).

Plant conditions are as follows:

- Reactor water level is -150", trending down.
- Only 1 SRV can be opened.
- Reactor pressure is 1100 psig, trending up.

The CRS has determined that Alternate RPV Depressurization Systems must be used to depressurize the RPV.

Which of the following describes the ability to use the Main Steam Line Drains as an alternate RPV Depressurization System?

Main Steam Line Drains are...

A.	closed and require bypassing interlocks to open.
B.	closed, but can be opened without bypassing interlocks.
C.	open and already lined up as an alternate RPV Depressurization System.
D.	open, but additional action is required to line them up as an alternate RPV Depressurization System.

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per CPS 4411.09 RPV PRESSURE CONTROL SOURCES section 2.2.1.11 Main Steam Line Drains, the following MSL Inboard Drain valves are opened as necessary: • 1B21-F016, MS Drn & MSIV Byp Inbd Isol Valve • 1B21-F019, MS Drn & MSIV Byp Outbd Isol Valve • 1B21-F020, MSIV Byp Vlv For MS Line Warm Up (normally closed; used to align flowpath to the downstream MSL drains) • 1B21-F021, Inbd MSIV Before Seat Warmup Drn Vlv (in parallel with F033) • 1B21-F033, Inbd MSIV Before Seat Warmup Drn Vlv (in parallel with F033) Per CPS 4001.02C001 Automatic Isolation Checklist, RPV Level Low - Level 1 (-145.5 in) will cause a Group 1 isolation of 1B21-F022A-D, 28A-D, 67A-D, **F016 and F019**. Per CPS 3101.01E001 Main Steam Electrical Lineup, 1B21-F021 is normally deenergized (Multiple Spurious Operation (MSO) related breaker). Per CPS 3005.01 Unit Power Changes, steps 8.1.9 and 8.2.11 direct verifying 1B21-F033 closes when > 42% RTP and opens when < 42% power. Therefore, with RPV level at -150", Group 1 Isolation interlocks will have to be bypassed per CPS 4411.09 RPV PRESSURE CONTROL SOURCES step 2.2.1.1 to allow opening of 1B21-F016 and F019 to use the MSL drains as an alternate depressurization system. Incorrect responses: B is incorrect but plausible. This answer would be correct if 1B21-F016 and F019 were normally closed at rated thermal power (similar to the normal positions for 1B21-F068 Outbd MSIV Before Seat Warmup Drn Vlv, F069 Outbd MSIV Before Seat Norm Drn Vlv, F070 MS Low Point Warm Up Drn Vlv, and F071 MS Low Point Normal Drn Vlv) and if RPV Level was > -145.5 inches. C is incorrect but plausible. This answer would be correct if RPV level was > -145.5 inches. 1B21-F016 and F019 are normally open valves, and F033 automatically opens when < 42% power. D is incorrect but plausible. This answer is partially correct in that turning the breaker on for 1B21-F021 (MSO valve) is required to open the valve and would be fully correct if RPV level > -145.5 inches.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295025.EA1.01	EA1.01	2.9	3.0	1		3

System Name
High Reactor Pressure

Category Statement
Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.7 / 45.6)

K/A Statement
Main steam line drains

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q12 295025 A1.01

Other NRC Data

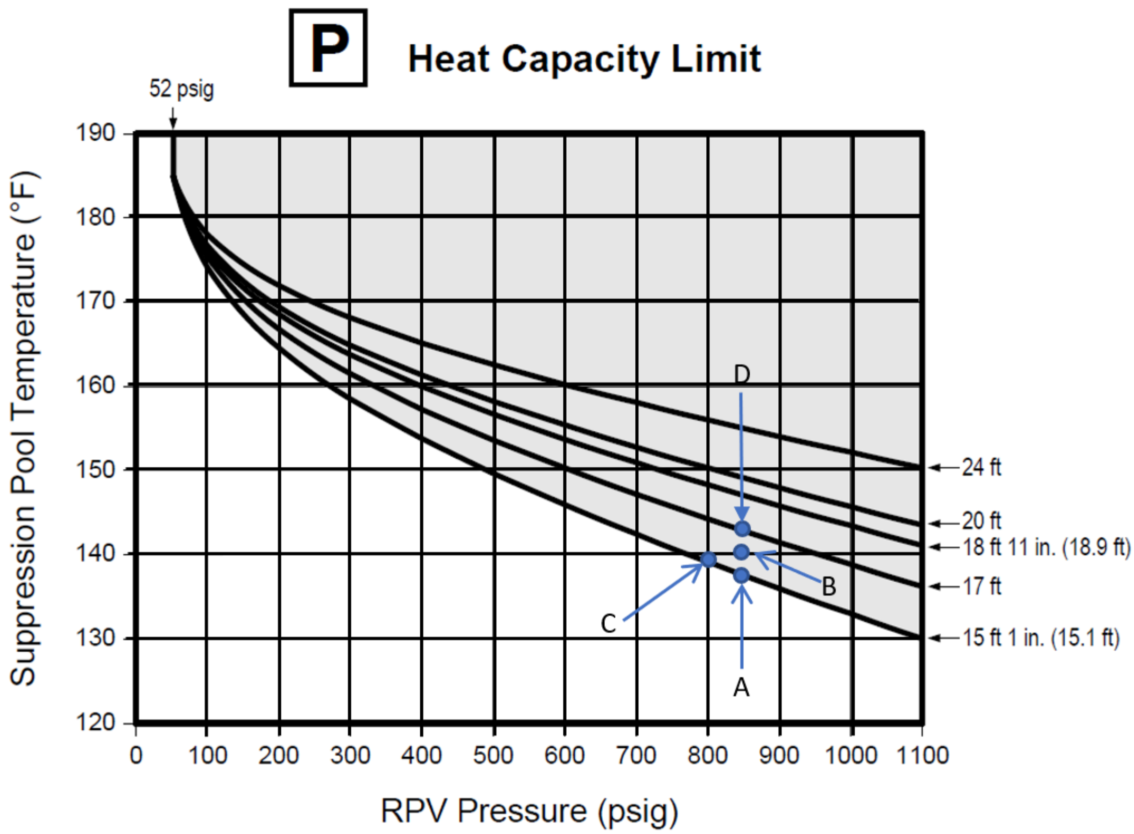
References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to operate the MSL Drains IAW pressure control procedures to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall facts from a procedure and knowledge of system interlocks to answer the question (1-F/1-I).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3005.01 Rev. 43f • CPS 3101.01E001 Rev. 16 • CPS 4001.02C001 Rev. 16b • CPS 4411.09 Rev. 6a		
Training Objective	LP87594.01.12 Given CPS 4411.09 explain notes, cautions, conditional requirements and/or describe actions taken for RPV Pressure Control Sources for: .01 Main Steam		
Previous NRC Exam Use	None		

13	ID: 2104683	Points: 1.00

Plant conditions are as follows:

- Suppression Pool Level is 16.05 feet.
- Reactor Pressure is 850 psig.

The Heat Capacity limit for these conditions is first met at point...



A.	A
B.	B
C.	C
D.	D

Answer:	A

Answer Justification / Plausibility Statements
A is correct. Interpolating between pool level curves is not allowed. With SP level at 16.05 ft, the lower curve (15.1 ft.) must be used. The intersection of 850 psig and the 15.1 ft. suppression pool level curve is ~137°F (point A). Incorrect Responses: B is incorrect but plausible. This answer would be correct if interpolating between pool level curves was allowed. The intersection of 850 psig and midway between the 15.1 ft. and 17 ft. suppression pool level curve is ~140°F (point B). C is incorrect but plausible. This answer would be correct if interpolation of RPV pressure was not allowed (similar to the restrictions on interpolating the SP level curves). The intersection of 800 psig and the 15.1 ft. suppression pool level curve is ~139°F (point C). D is incorrect but plausible. This answer would be correct with suppression pool level at 17 ft. and RPV pressure at 850 psig. The intersection of the 850 psig and the 17 ft. suppression pool level curve is ~143°F (point D).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295026.EA2.02	EA2.02	3.8	3.9	1		5

System Name
Suppression Pool High Water Temperature

Category Statement
Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
Suppression pool level

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q13 295026 A2.02

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to interpret the suppression pool level parameter provided in the stem to determine when the heat capacity limit has been reached to answer the question.		
SRO-Only Justification	N/A		
Additional Information	Cog Level justification - this is a high cog question written at the analysis and comprehension level. The examinee has to evaluate parameters using a graph to answer the question (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 4402.01 Rev. 30		
Training Objective	LP87558.01.08 Given a diagram of EOP-6, explain the use and/or function of the following inserts: .08 Figure P, Heat Capacity Limit		
Previous NRC Exam Use	None		

14	ID: 2104696	Points: 1.00

The plant is operating at rated thermal power.

A reactor coolant leak is causing Containment temperature to rise.

Annunciator 5004-3F SPDS CSF ALARM will be received when Containment temperature first reaches _____(1)_____°F.

A scram is required when containment temperature reaches a maximum of _____(2)_____°F.

A.	(1) 122 (2) 184
B.	(1) 122 (2) 329
C.	(1) 150 (2) 184
D.	(1) 150 (2) 329

Answer:	A

Answer Justification / Plausibility Statements

A is correct:

Per CPS 5004.03 (3F) SPDS CSF ALARM, the alarm setpoint for Containment temperature is 122°F which corresponds to the EOP-6 Primary Containment Control entry condition on Containment temperature.

Per EOP-6 Primary Containment Control, the containment temperature leg requires a scram to be inserted before containment temperature reaches 185°F.

Incorrect responses:

B is incorrect but plausible. Part 1 is correct. Part 2 (329°F) is plausible because EOP-6 requires a scram to be inserted before Drywell temperature reaches 330°F.

C is incorrect but plausible. 150°F is the EOP-6 entry condition and SPDS CSF Alarm setpoint for high drywell temperature. Part 2 is correct.

D is incorrect but plausible:

- Part 1 - 150°F is the EOP-6 entry condition and SPDS CSF Alarm setpoint for high Drywell temperature.
- Part 2 - EOP-6 requires a scram to be inserted before Drywell temperature reaches 330°F.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295027	B2.4.01	4.6	4.8	1	1	N/A

System Name
High Containment Temperature (Mark III Containment Only)

Category Statement
Knowledge of EOP entry conditions and immediate action steps. (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

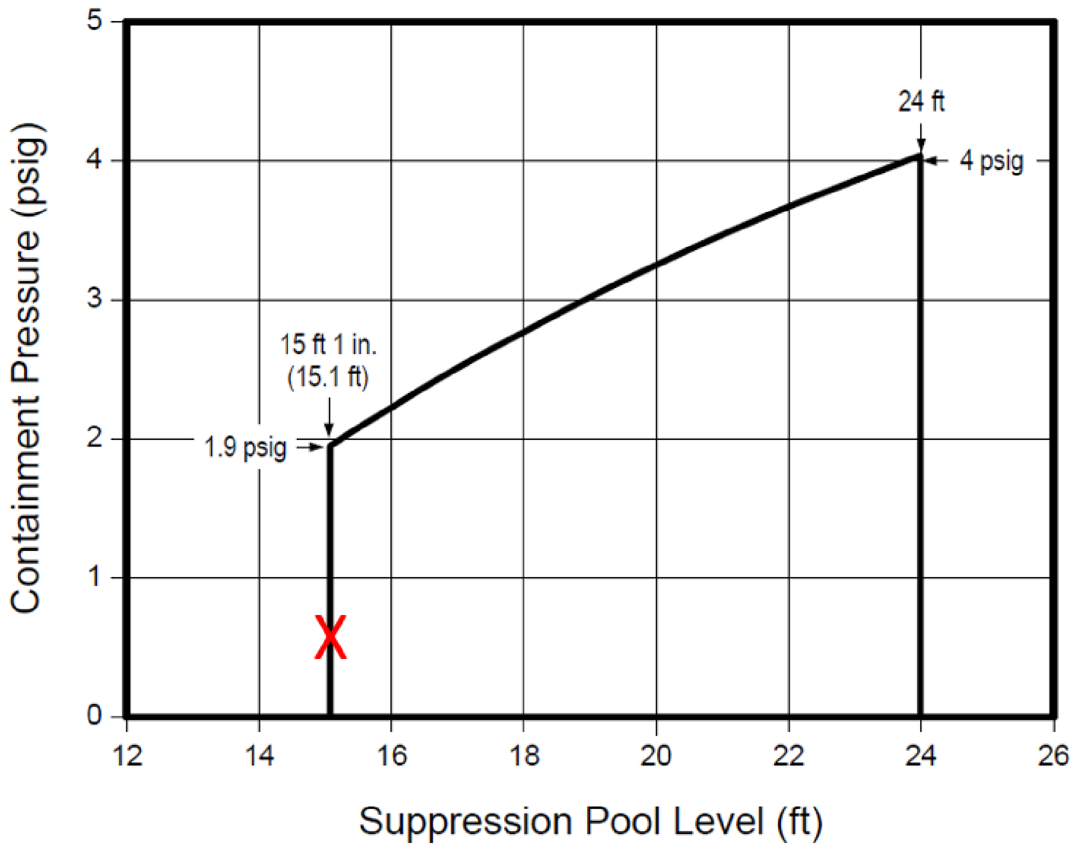
	Q14 295027 2.4.1

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of the EOP-6 entry condition for high containment temperature and actions that are required to be taken to answer the question.		
SRO-Only Justification	N/A		
Additional Information	Cog Level Justification - this is a low cog question written at the memory level - requires recall of procedure facts to answer the question (1-F).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5004.03 (3F) Rev. 28b • CPS 4402.01 Rev. 30		
Training Objective	LP87558.01.01 From memory, state the plant conditions which require entry/reentry to EOP-6.		
Previous NRC Exam Use	None		

15	ID: 2104743	Points: 1.00

The plant was operating at rated thermal power when a transient occurred resulting in suppression pool level and containment pressure parameters as indicated by the red 'X' on Figure N below.



Actions are required to prevent...

A.	uncovering the RCIC Turbine exhaust sparger.
B.	damage to the low pressure ECCS pumps due to inadequate NPSH.
C.	incomplete condensing of the steam discharged through the horizontal vents during a LOCA.
D.	opening a path directly into the containment from the Combustible Gas Control System (HG) Mixing Compressor discharge lines.

Answer:	C

Answer Justification / Plausibility Statements
C is correct: Per the EOP-TB, Line 4 (the vertical line at 15'1" SP level) is the suppression pool level corresponding to an elevation two feet above the horizontal vents. If suppression pool level is below this elevation, the RPV may not be kept in a pressurized state since steam discharged through the vents may not be condensed. Incorrect responses: A is incorrect but plausible. Per M05-1079-1 at B-7, the RCIC Turbine does exhaust to the suppression pool via line 1RI08C. The top of the exhaust sparger begins at elevation 730' 11". This corresponds to a suppression pool level of 18' 11". At 15'1", the RCIC Exhaust Sparger will be uncovered, however there are no limitations on suppression pool level based on uncovering the RCIC Turbine Exhaust Sparger. B is incorrect but plausible. EOP-6 states that low suppression pool level affects the margin to NPSH / Vortex Limits (Detail Z). For low pressure ECCS , the minimum suppression pool level limit is 11 ft. The EOP tech bases states that the NPSH and vortex restrictions are specified in a caution rather than explicit operating limits to provide necessary event-specific flexibility and to avoid potential conflicts between parallel parameter control paths. D is incorrect but plausible. Per the EOP-TB, Mixer (HG) operation is permitted in the Drywell/Containment Pressure and Drywell/Containment Hydrogen branches (Parts L and Q) only if suppression pool level is above 13 ft. 1 in. (13.1 ft.). If suppression pool level were below 13 ft. 1 in. (13.1 ft.), the mixer discharge would be uncovered and mixer operation would open a path from the drywell directly into the containment. A break in the drywell could then overpressurize the containment. The direction to stop mixers in the Suppression Pool Level branch ensures that appropriate action is taken if suppression pool level drops below the discharge elevation after the mixers are started.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295030.EK1.01	EK1.01	3.8*	4.1*	1		5

System Name
Low Suppression Pool Water Level

Category Statement
Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.8 to 41.10)

K/A Statement
Steam condensation

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10
41.8	41.8
41.9	41.9

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q15 295030 K1.01

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of the reason for required actions during a low suppression pool water level transient to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The candidate has to analyze conditions in the stem and then determine reasons for required actions based on the analysis (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• EOP-TB Rev. 7 • CPS 4402.01 Rev. 30		
Training Objective	N-CL-OPS-DB-LP87558.01.02 Given a diagram of EOP-6: .02 State the bases for each individual step/action of EOP-6		
Previous NRC Exam Use	None		

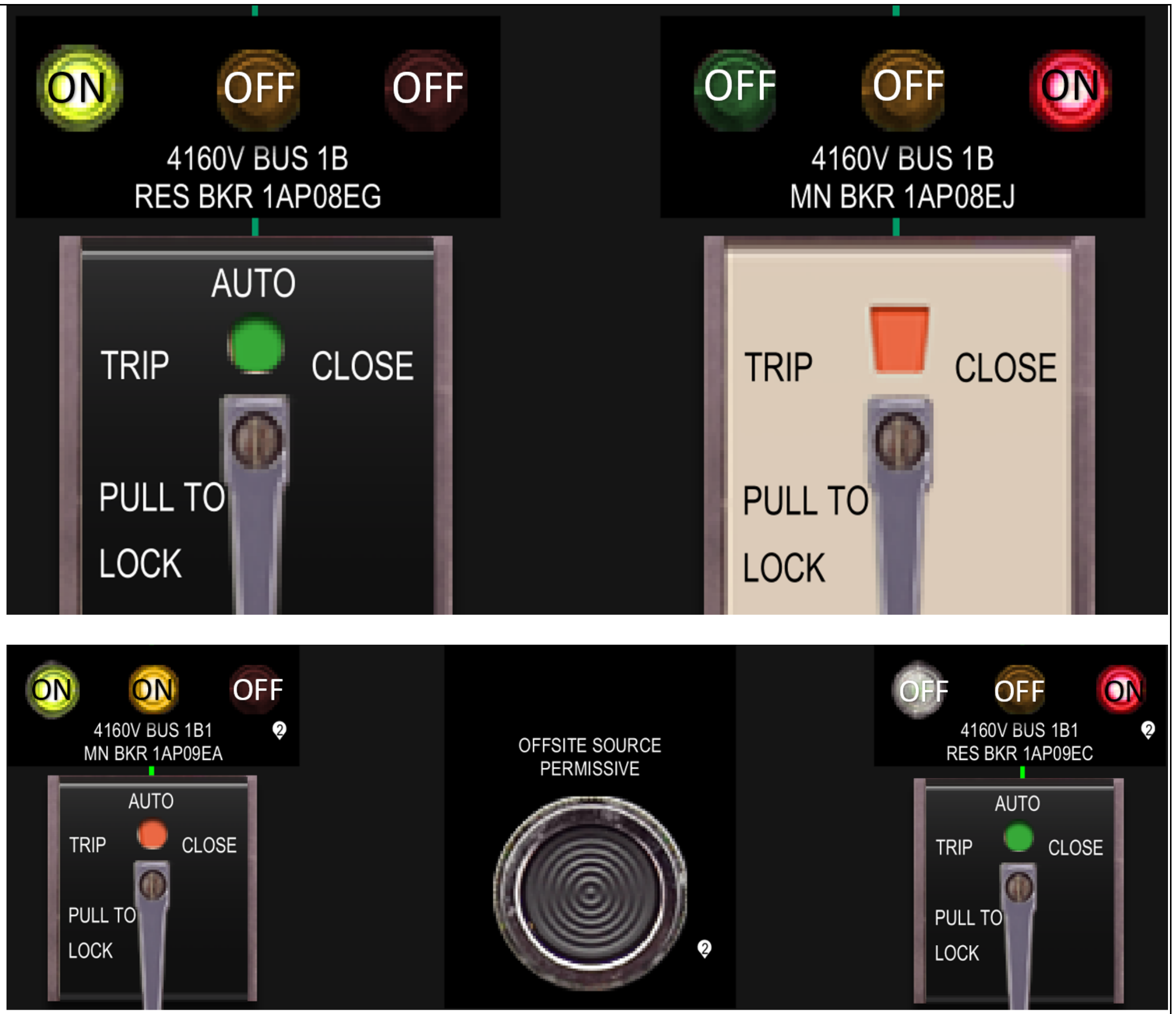
16	ID: 2104746	Points: 1.00

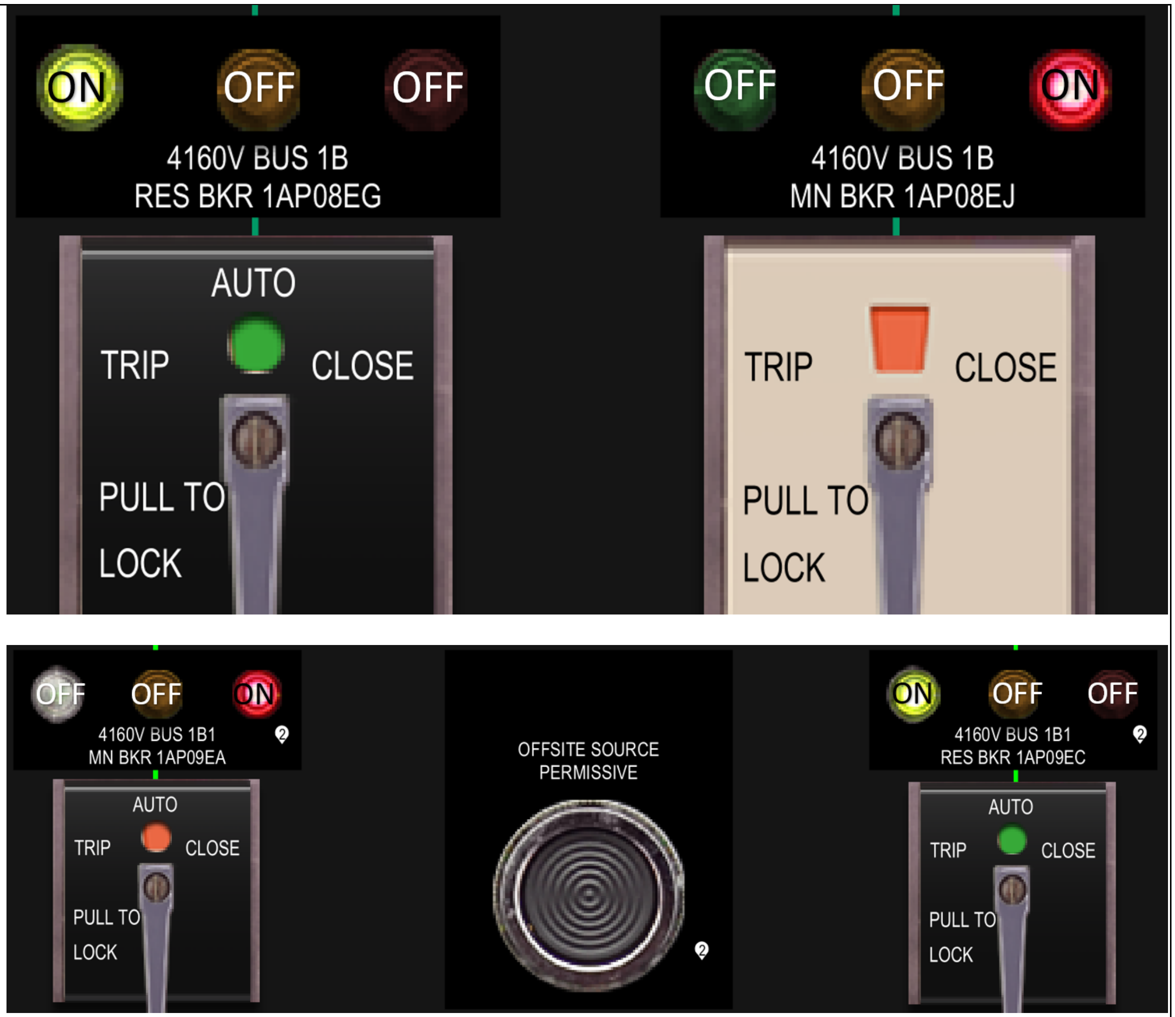
The plant was operating at rated thermal power with AC distribution aligned as follows:

- 4160V Bus 1B powered from its normal source.
- 4160V Bus 1B1 powered from the Reserve Auxiliary Transformer (RAT).

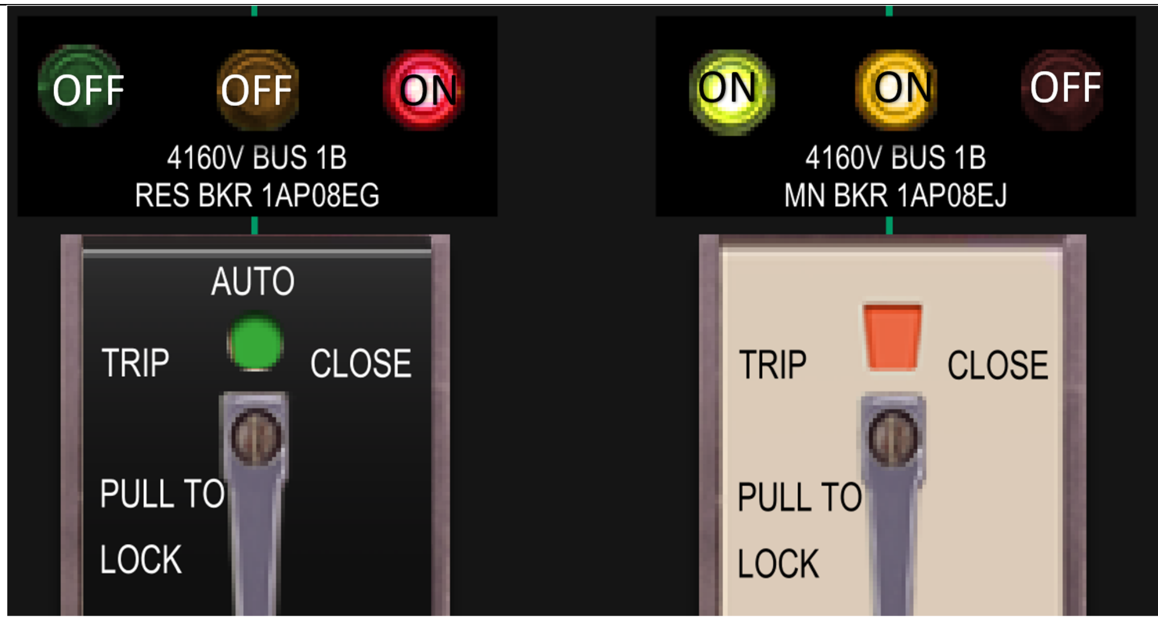
THEN, a transient resulted in RPV water level lowering to -150".

Which of the graphics below show the expected response of the AC distribution system 2 minutes following the RPV level transient?

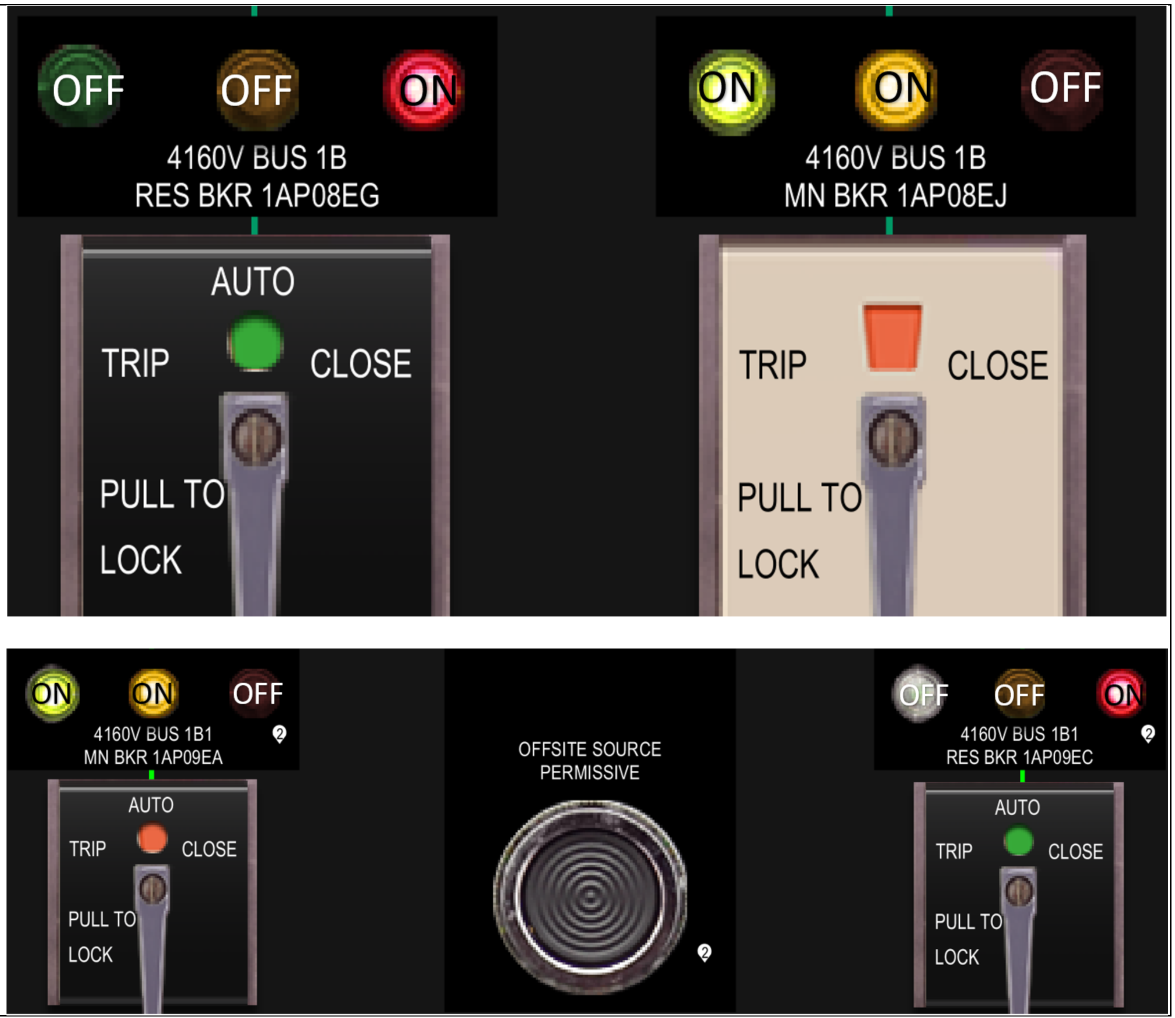
1	

2	

ON OFF OFF
4160V BUS 1B
RES BKR 1AP08EG
AUTO
TRIP CLOSE
PULL TO
LOCK
OFF OFF ON
4160V BUS 1B
MN BKR 1AP08EJ
TRIP CLOSE
PULL TO
LOCK
OFF OFF ON
4160V BUS 1B1
MN BKR 1AP09EA
AUTO
TRIP CLOSE
PULL TO
LOCK
OFFSITE SOURCE
PERMISSIVE
ON OFF OFF
4160V BUS 1B1
RES BKR 1AP09EC
AUTO
TRIP CLOSE
PULL TO
LOCK

3	 OFF OFF ON 4160V BUS 1B RES BKR 1AP08EG AUTO TRIP CLOSE PULL TO LOCK ON ON OFF 4160V BUS 1B MN BKR 1AP08EJ TRIP CLOSE PULL TO LOCK  OFF OFF ON 4160V BUS 1B1 MN BKR 1AP09EA AUTO TRIP CLOSE PULL TO LOCK OFFSITE SOURCE PERMISSIVE ON OFF OFF 4160V BUS 1B1 RES BKR 1AP09EC AUTO TRIP CLOSE PULL TO LOCK

4



OFF
OFF
ON
4160V BUS 1B
RES BKR 1AP08EG
AUTO
TRIP
CLOSE
PULL TO
LOCK
ON
ON
OFF
4160V BUS 1B
MN BKR 1AP08EJ
TRIP
CLOSE
PULL TO
LOCK
ON
ON
OFF
4160V BUS 1B1
MN BKR 1AP09EA
AUTO
TRIP
CLOSE
PULL TO
LOCK
OFFSITE SOURCE
PERMISSIVE
OFF
OFF
ON
4160V BUS 1B1
RES BKR 1AP09EC
AUTO
TRIP
CLOSE
PULL TO
LOCK

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

Answer:	C

Answer Justification / Plausibility Statements

C is correct:
 	
Initial conditions are as follows:
 	
• 4160V Bus 1B Main Feed Breaker (from the 1B Unit Auxiliary Transformer) is closed.
• 4160V Bus 1B Reserve Feed Breaker (from Reserve Auxiliary Transformer 'C') is open.
• 4160V Bus 1B1 Main Feed Breaker (from Reserve Auxiliary Transformer 'B') is closed.
• 4160V Bus 1B1 Reserve Feed Breaker (from the Emergency Reserve Auxiliary Transformer) is open.
• Division 2 Diesel Generator is in standby.
 	
A level transient to -150" will result in the following sequence of events:
 	
• Reactor Scram at Level 3 (+8.9") (5004-1B)
• Generator Reverse Power Trip (5008-2E/2G)
• Auto transfer to the 4160V Bus 1B Reserve Feed source (5012-2A). The Reserve Feed breaker for 4160V Bus 1B will automatically close and the Main Feed breaker will trip.
• Automatic start of the Div 2 Diesel Generator (ITS B3.8.1 pages B3.8-1 and B3.8-2).
    • The Div 2 DG output breaker will remain open.
    • The 4160V Bus 1B1 Main Feed Breaker will remain closed, and the 1B1 Reserve Feed Breaker will remain open.
 	
These actuations are represented in Graphic 3.
 	
Incorrect responses:
 	
A is incorrect but plausible. Graphic 1 is correct for a Level 1 undervoltage condition on 4160V Bus 1B1, which will cause 4160V Bus 1B1 to automatically transfer to the reserve feed source.
 	
B is incorrect but plausible. Graphic 2 is the pre-event alignment and is plausible because the RPV level transient does not result in a direct trip of the 4160V Bus 1B Main Feed Breaker. The breaker trip is caused by the sequence of events listed above.
 	
D is incorrect but plausible. A level 1 undervoltage signal will cause the 4160V Bus 1B1 Main Feed Breaker to trip and the Reserve Feed Breaker to automatically close. The RPV Level 1 actuations will cause the Div 2 DG to automatically start, but does not cause the Main and Reserve feed sources to shift.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295031.EK2.15	EK2.15	3.2	3.2	1		2

System Name
Reactor Low Water Level

Category Statement
Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: (CFR: 41.7 / 45.8)

K/A Statement
A.C. distribution: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q16 295031 K2.15

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of how the AC Distribution system responds to low RPV level events to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This question is a high cog question written at the analysis and comprehension level. The examinee has to analyze 4 graphics and determine expected response to the conditions in the stem to answer the question (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5004.01 (1B) Rev. 28c • CPS 5008.02 (2E) Rev. 27c • CPS 5012.02 (2A) Rev. 23 • ITS B3.8.1 (B3.8-1) Rev. 3-5 • ITS B3.8.1 (B3.8-2) Rev. 0		
Training Objective	262001.05 Discuss the Auxiliary Power system automatic functions/interlocks including purpose, signals, set points, sensing points, when bypassed, how/when they are. .3 4.16 kV Normal and Reserve Breaker Control .4 4.16 kV Bus Automatic Transfer 264000.15 Given DIESEL GENERATOR/DIESEL FUEL OIL System initial conditions, PREDICT how the system and/or plant parameters will respond to the manipulation of the following controls. .6 LOCA signal		
Previous NRC Exam Use	None		

17	ID: 2104768	Points: 1.00

The plant was operating at rated thermal power when a scram occurred.

Immediate operator actions for the reactor scram have been completed.

Current plant conditions are as follows:

Parameter	Value	Trend
Rx Power	25%	stable
RPV pressure	917 psig	stable
RPV level	25"	stable

Standby Liquid Control (SLC) pumps are started ___(1)___.

SLC is injected under these conditions to ___(2)___.

A.	(1) with NO direction required from the CRS (2) protect the containment
B.	(1) with NO direction required from the CRS (2) preclude power oscillations
C.	(1) ONLY when directed by the CRS (2) protect the containment
D.	(1) ONLY when directed by the CRS (2) preclude power oscillations

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per the EOP Tech Bases for EOP-1A ATWS RPV Control, if reactor power remains above the APRM downscale setpoint following multiple attempts to scram the reactor, Clinton operating practices call for immediate injection of boron to preclude power oscillations and ensure that the plant remains in a controlled state. Per OP-CL-101-111-1001-F-02 Start of Scram Choreography (ATWS), the 'B' RO is directed to initiate SLC under the following conditions: • manual scram and ARI have been initiated, and • power remains greater than 5%. Incorrect Responses: A is incorrect but plausible. The first part is correct. The second part would be correct if heat was being added to the containment. With reactor pressure stable at 917 psig and power at 25% (below the capacity of the Main Turbine Bypass valves), heat is <u>not</u> being added to the Suppression Pool and containment integrity is not being jeopardized. C is incorrect but plausible. This answer would be correct if: • the stem conditions presented were for a low power ATWS (power below 5%), and • heat was being added to the containment. With reactor pressure stable at 917 psig and power at 25% (below the capacity of the Main Turbine Bypass valves), heat is <u>not</u> being added to the Suppression Pool and containment integrity is not being jeopardized. D is incorrect but plausible. This answer would be correct if the stem conditions presented were for a low power ATWS (power below 5%) where SLC Pumps are started based on EOP-1A ATWS RPV Control conditions at the direction of the CRS. The second part is correct.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295037.EK3.02	EK3.02	4.3*	4.5*	1		1

System Name
SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

Category Statement
Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : (CFR: 41.5 / 45.6)

K/A Statement
SBLC injection

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q17 295037 K3.02

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the reasons for injecting SLC during an ATWS to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to analyze parameters in the stem and then determine required actions to answer the question (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• OP-CL-101-111-1001-F-02 Rev. 1a • OP-CL-102-106-1001 Rev. 8c • EOP-TB (5-53) Rev. 7		
Training Objective	N-CP-OPS-DB-LP87553.01.04 State the boron injection methods and flowpaths which can be used to respond to an ATWS.		
Previous NRC Exam Use	None		

18	ID: 2029210	Points: 1.00

The Main Control Room (MCR) Ventilation (VC) System is currently aligned in the NORMAL Mode with the 'A' train in operation and the 'B' train in standby. Cont Rm Trn Min OS Air Dmpr 0VC01YA (East) and 0VC01YB (West) are OPEN.

Then, a high off-site release results in the receipt of the following annunciators:

- HI RADIATION CONT RM HVAC SYST DIVISION 1 (5050-7M)
- HI RADIATION CONT RM HVAC SYST DIVISION 2 (5052-7M)

The Outside Air Inlet Rad Monitors for the MCR VC System on P801-66B and 67B indicate the following:

- 1RIX-PR009A (West): 12 mr/hr
- 1RIX-PR009B (West): 14 mr/hr
- 1RIX-PR009C (East): 1 mr/hr
- 1RIX-PR009D (East): 2 mr/hr

The operating VC Train Makeup Air Fan (0VC05CA) ____(1)______.

The Cont Rm Trn Min OS Air Dmpr (0VC01YB) ____(2)____.

A.	(1) remains in standby (2) must be manually SHUT
B.	(1) automatically starts (2) must be manually SHUT
C.	(1) remains in standby (2) will automatically SHUT
D.	(1) automatically starts (2) will automatically SHUT

Answer:	B

Answer Justification / Plausibility Statements
B is correct. IAW CPS 5050.07 and CPS 5052.07 AUTO ACTIONS, with the detectors (1RIX-PR009A/B/C/D) alarming (>10 mR/hr or Downscale) in the provided combination of channels A and B then the operating MCR HVAC train will realign to the High Radiation Isolation mode as follows: • 0VC09YA(B), Sply Air Trn A(B) Filt Inlet Dmpr opens. • 0VC10YA(B), Sply Air Trn A(B) Filt Byp Dmpr closes. • 0VC11YA(B), Sply Air Trn A(B) Filt Outlet Dmpr opens. During normal operations of the VC system, 0VC05CA Cont Rm HVAC A MU Air Fan is in standby. 0VC01YA/B Cont Rm Trn A/B Min OS Air Dmprs are open. Per CPS 3402.01 Control Room HVAC (VC) section 8.3.3 High Radiation Isolation: • Step 8.3.3.4, verify running/start 0VC05CA, Cont Rm HVAC A MU Air Fan • Step 8.3.3.8, requires the minimum air damper with the lowest radiation level to be opened/verified open, and the damper on the other side to be closed. Incorrect Responses: A is incorrect but plausible. This answer would be correct if the trips were on channels A & D. C is incorrect but plausible. This answer would be correct if: • the trips were on channels A & D, and • the 0VC01YB received an auto closure signal similar to 0VC10YA(B), Sply Air Trn A(B) Filt Byp Dmpr. D is incorrect but plausible. This answer would be correct if the 0VC01YB received an auto closure signal similar to 0VC10YA(B), Sply Air Trn A(B) Filt Byp Dmpr.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295038.EA1.07	EA1.07	3.6	3.8	1		9

System Name
High Off-Site Release Rate

Category Statement
Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE : (CFR: 41.7 / 45.6)

K/A Statement
Control room ventilation: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q36 290003 A4.02 X

	Q18 295038 A1.07

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee must demonstrate the ability to operate and monitor the Control Room Ventilation System during a high off-site release condition to answer the question. Per ITS B3.3.7.1 (B3.3-217), the Control Room Air Intake Radiation Monitors measure radiation levels exterior to the inlet ducting of the MCR. A high off-site release indicated by high radiation levels at the MCR Air Intake Radiation Monitors will automatically initiate the CRV System high radiation mode.		
SRO Only Justification Statement:	N/A		
Additional Information:	Question is written at the memory level (low cog). The examinee has to recall the VC system automatic actions and manual actions necessary based on conditions presented in the stem. (1-P)		
NRC Exams Only (as applicable)			
Question Type:	Bank (CL-ILT-2029210)	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 5050.07 (7M) Rev. 33b • CPS 5052.07 (7M) Rev. 34 • CPS 3402.01 Rev. 32c • ITS B3.3.7.1 (B3.3-217)		
Training Objective:	290003.05 Discuss the CONTROL ROOM HVAC system automatic functions/interlocks including purpose, signals, set points, sensing points, when bypassed, how/when they are. .1 High Radiation Initiation Actuation		
Previous NRC Exam Use:	None		

19	ID: 2107243	Points: 1.00

The plant was operating at rated thermal power when the Reserve Auxiliary Transformer (RAT) Automatic Deluge System (ADS) was manually initiated due to a fire in the vicinity of a Reserve Auxiliary Transformer.

The following alarm was received on 1H13-P841:

- DEVICE 24-19 IN ALARM - RAT ADS

As a direct result of the RAT ADS suppression system activation, the __________ will trip.

A.	RAT Circuit Switcher (4538)
B.	RAT 'B' Cooling Fans and Pumps ONLY
C.	RAT SVC Breakers 1AP103E and 1AP104E
D.	RAT 'A', 'B', and 'C' Cooling Fans and Pumps

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per CPS 5121 (5121-2419A), initiation of the Reserve Auxiliary Transformer (RAT) Automatic Deluge System (ADS) will cause the RAT 'B' Cooling Fans and Pumps to trip. Per CPS 9071.10C020 RAT 'B' ADS AND NON SEG BUS DUCT AT UAT SYSTEM PANEL 1FP14JB AND 1FP16J SIMULATED AUTOMATIC ACTUATION CHECKLIST, note before step 8.2.1 states that only RAT 'B' is serviced by RAT ADS. RAT 'A' and RAT 'C' are not ADS equipped. Incorrect Responses: A is incorrect but plausible. Per CPS 5010.01 (1A), RAT 1 Circuit Switcher 4538 will trip on the following signals: • Fault in the RAT A(B)[C] • Fault on Bus 1RT6 • Fault on Bus 1RT4 • Fault on Bus 1RTC4 • Manual opening of the RAT circuit switcher • North Bus Differential Overcurrent Tripping 4538 will disconnect 345KV SY power to all three RAT's, but is not actuated by fire suppression actuation. C is incorrect but plausible. Per 5121-1313A DEVICE 13-13 IN ALARM, a <u>RAT SVC Bldg 737' fire</u> will cause the RAT SVC unit to shut down, the SVC yard breaker to trip open, the SVC building HVAC system to shutdown and the FM-200 fire suppression system to actuate. D is incorrect but plausible. Original plant design incorporated a single RAT transformer that was equipped with an Automatic Deluge sprinkler system. When the AP design was changed to split the RAT transformers into 3 separate transformers, ADS was installed on RAT 'B' alone.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
600000.AA2.14	AA2.14	3.0	3.6	1		8

System Name
Plant Fire On Site

Category Statement
Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE:

K/A Statement
Equipment that will be affected by fire suppression activities in each zone

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q19 600000 A2.14

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to determine the equipment that will be affected by initiation of the RAT ADS fire suppression system to answer the question.
SRO-Only Justification	N/A
Additional Information	This is a low cog question written at the memory level. The examinee has to recall facts about a system to answer the question (1-F).
NRC Exams Only	
Question Type	New — **Difficulty** N/A
Technical Reference and Revision #	• CPS 5121-2419A Rev. 27d • CPS 9071.10C020 Rev. 0a • CPS 5121-1313A Rev. 27d
Training Objective	286000.03 DESCRIBE the function, operation, interlocks, trips, physical location, and power supplies of the following FP FD FIRE PROTECTION - DETECTION System components. .4 Deluge system .7 Transformer Cooling Fan-Deluge System Interlock
Previous NRC Exam Use	None

20	ID: 2104785	Points: 1.00

A degraded grid condition exists. Attempts to raise the grid voltage by raising MVAR load on the generator have resulted in the following conditions:

- Generator MVA load is 1200 MVA
- Generator Terminal Voltage is 20,460 Volts

Given the following table from CPS 3105.05 Generator (TG) Section 8.5.2 Abnormal Voltage,

Voltage/(%)	LAG MVA Limit	LEAD MVA Limit
20,900 / (95.0%)	1265	1265
20,790 / (94.5%)	1195	1129
20,680 / (94.0%)	1189	1118
20,570 / (93.5%)	1182	1106
20,460 / (93.0%)	1176	1094
20,350 / (92.5%)	1170	1082
20,240 / (92.0%)	1164	1070
20,130 / (91.5%)	1157	1059
20,020 / (91.0%)	1151	1047
19,910 / (90.5%)	1144	1036
19,800 / (90.0%)	1138	1025
19,580 / (89.0%)	1125	1002
19,360 / (88.0%)	1113	980
19,140 / (87.0%)	1100	957
18,920 / (86.0%)	1088	936
18,700 / (85.0%)	1075	914

Determine which one of the following:

(1) identifies the target MVA load, and

(2) describes the APPROPRIATE method to reduce the MVA load?

A.	(1) 1176 MVA (2) Lower Reactor Recirc flow.

B.	(1) 1094 MVA (2) Lower Reactor Recirc flow.

C.	(1) 1176 MVA (2) Lower generator excitation.

D.	(1) 1094 MVA (2) Lower generator excitation.

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per CPS 3105.05 Generator (TG) Section 8.5.2 Abnormal Voltage, for operation below 95% of rated voltage (20,900), the operator must reduce MVA load to a value listed in the table based on the given terminal voltage. Additionally, since a LAGGING power factor is the normal operating state, the LAG MVA Limit of 1176 would be selected. CPS 3105.05 Section 8.5.2 Abnormal Voltage also provides the guidance the generator output (MVA) should be reduced "by lowering reactor power (only way to lower generator MVA)". Incorrect Responses: B is incorrect but plausible. This answer would be correct if the generator was operating at a LEADING power factor. C is incorrect but plausible. This answer would be correct if lowering generator voltage lowered only MVAR without affecting the MVA limit. D is incorrect but plausible. This answer would be correct if: • the generator was operating at a LEADING power factor, and • lowering generator voltage lowered only MVAR without affecting the MVA limit.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.700000	B2.1.07	4.4	4.7	3	N/A	N/A

System Name
Generator Voltage and Electric Grid Disturbances

Category Statement
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)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q50 700000 A1.04 V

	Q20 700000 2.1.7

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee demonstrate the ability to evaluate conditions in the stem and then make operational judgements during an electric grid disturbance to answer the question.
SRO-Only Justification	N/A
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to analyze the conditions in the stem and then determine required actions based on that analysis (3-SPK).
NRC Exams Only	
Question Type	Bank (CL-ILT-2029141) — **Difficulty** N/A
Technical Reference and Revision #	CPS 3105.05 Rev. 22d
Training Objective	245002.11 EVALUATE given key MAIN GENERATOR System parameters, if needed DETERMINE a course of action to correct or mitigate the following abnormal condition(s): .1 Trip Signal .2 Isolation Signal .3 Auto Start Signal .4 Condition/Lineup
Previous NRC Exam Use	None

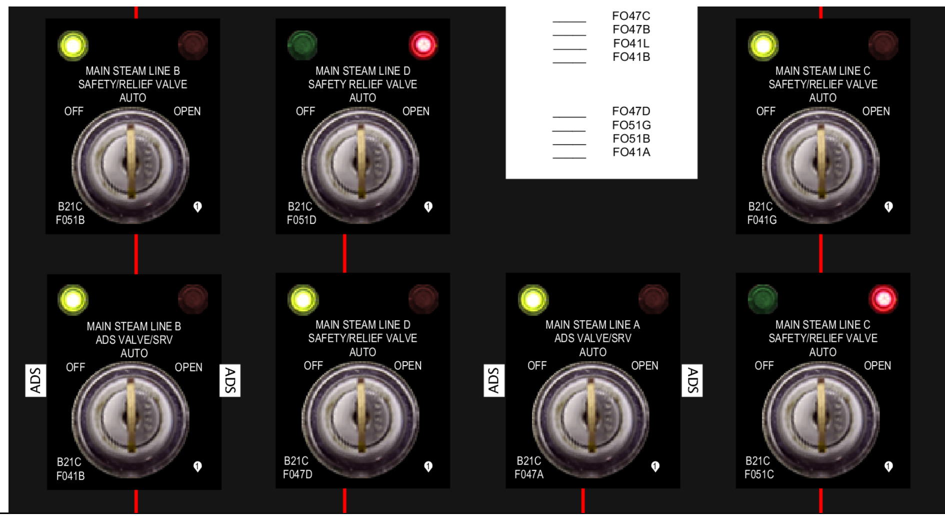
21	ID: 2104809	Points: 1.00

The plant was operating at rated thermal power when a Group 1 isolation occurred, resulting in a reactor scram.

Which of the following scenarios represent normal response to decay heat generation two minutes following the scram?

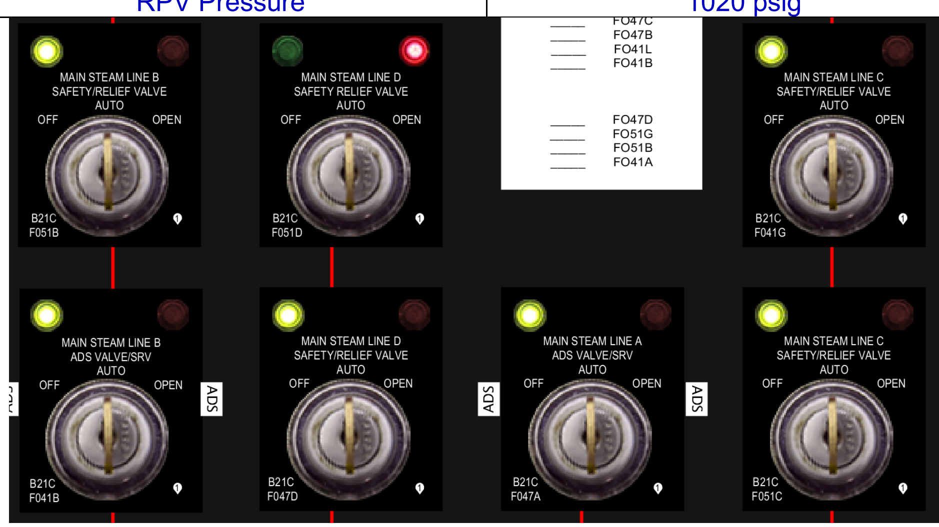
Scenario 1

Parameter	Value
RPV Pressure	950 psig

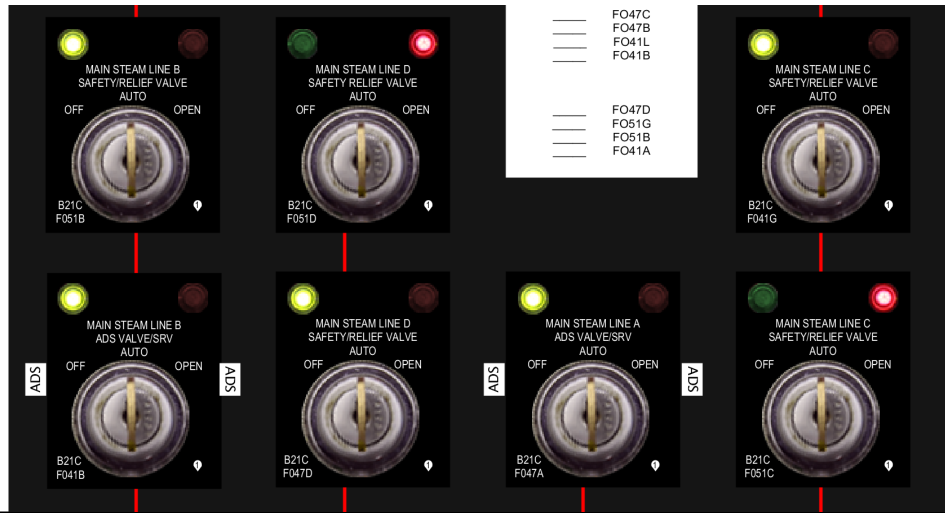


Scenario 2

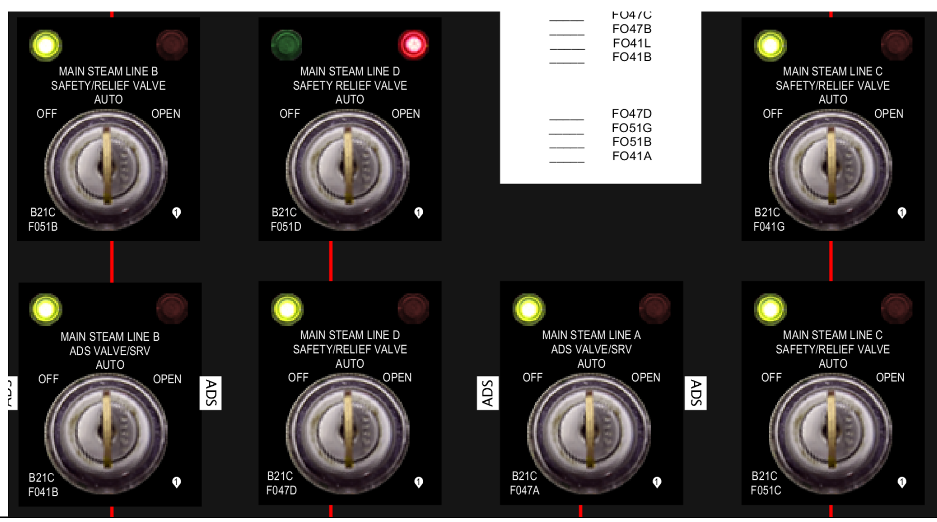
Parameter	Value
RPV Pressure	1020 psig



	Parameter	Value
Scenario 3	RPV Pressure	1110 psig



	Parameter	Value
Scenario 4	RPV Pressure	1090 psig



A.	Scenario 1
B.	Scenario 2
C.	Scenario 3
D.	Scenario 4

Answer:	B

Answer Justification / Plausibility Statements

B is correct:

Per CPS 5067.06 (6C), the relief and Low Low Set (LLS) setpoints for the 5 LLS SRVs are as follows:

SRV	Relief Setpoint (psig)	LLS Opening Setpoint (psig)	LLS Closure Setpoint (psig)
F051D	1103	1033	926
F051C	1113	1073	936
F051B	1113	1113	946
F047F	1113	1113	946
F051G	1113	1113	946

- Two minutes following the scram, decay heat generation will cause a single SRV (F051D) to cycle in relief mode at the low-low opening and closing setpoints.
- Per CPS 5067.06 (6C) Low-Low Setpt Div 1 Sealed In, the opening & closing setpoints for 1B21-F051D is 1033 psig and 926 psig. This is represented in scenario 2.

Incorrect Responses:

A is incorrect but plausible. This answer would be correct if the stem asked for the initial response of SRVs 51D and 51C immediately after the Group 1 isolation.

- The initial lift of SRV F051D at 1103 psig will initiate the low low set logic. Since reactor pressure is above the low low set opening setpoint for F051C (1073 psig), both 51D and 51C will initially actuate and will then reclose at 926 psig and 936 psig respectively. After the initial actuation, F051D will cycle between 926 psig and 1033 psig.

C is incorrect but plausible. This answer would be correct if the capacity of the SRVs was such that two SRVs were required to control pressure two minutes after the scram. Each SRV will pass ~ 6% of rated steam flow, which is lower than the decay heat generation rate two minutes following the scram.

D is incorrect but plausible. This answer would be correct if 51D lifted on subsequent lifts at 1103 psig (no setpoint setdown logic).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295007.AK1.02	AK1.02	3.1	3.4	1		3

System Name
High Reactor Pressure

Category Statement
Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : (CFR: 41.8 to 41.10)

K/A Statement
Decay heat generation

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10
41.8	41.8
41.9	41.9

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q21 295007 K1.02

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of the decay heat generation and the associated SRV response to a high pressure scram condition to answer the question.		
SRO-Only Justification	N/A		
Additional Information	Cog Level Justification - this is a high cog question written at the analysis and comprehension level. The examinee has to analyze 4 scenarios, which include graphic indications, and then determine which graphic represents a condition showing proper operation of a system (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-N14018)	**Difficulty**	N/A
Technical Reference and Revision #	CPS 5067.06 (6C) Rev. 31		
Training Objective	239001.03 DESCRIBE the function, operation, interlocks, trips, and power supplies of the following MAIN STEAM System components. .2 Safety/Relief Valves and Piping		
Previous NRC Exam Use	ILT 14-1 NRC Exam		

22	ID: 2104896	Points: 1.00

Which of the following describes the interrelationship between High Reactor Water Level and the Main Turbine?

The Main Turbine trip signal is generated by _____(1)_____ logic.

The Main Turbine trip _____(2)______ bypassed with the Reactor System Mode Switch (RMS) in START/HOT STBY position.

A.	(1) 2 out of 3 (2) is

B.	(1) 2 out of 3 (2) is NOT

C.	(1) 2 out of 4 (2) is

D.	(1) 2 out of 4 (2) is NOT

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per CPS 9538.08 FW Reactor Vessel Water Level 1C34N004C Channel Functional, step 2.1.3 states that Alarm Unit 1C34K624C actuates on high reactor water level (Level 8). Alarm unit provides trip signal to a Two-Out-of-Three logic circuit which, when satisfied, causes Main Turbine and all three Reactor Feed Pumps (RFPs) to trip. Per CPS 9538.08 FW Reactor Vessel Water Level 1C34N004C Channel Functional, step 5.1, the surveillance can be performed in any Mode of plant operation (NOT RMS dependent) with Reactor Level 8 condition NOT present. Incorrect Responses: A is incorrect but plausible. Part 1 is correct. Part 2 is plausible because the RPS trip on High Reactor Water Level is bypassed with any RMS position other than Run (CPS 3305.01 Appendix A). C is incorrect but plausible. The RPS trip on High Reactor Water Level is 2 out of 4 logic. D is incorrect but plausible. The RPS trip on High Reactor Water Level is 2 out of 4 logic and the RPS trip is bypassed with any RMS position other than Run (CPS 3305.01 Appendix A).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295008.AK2.08	AK2.08	3.4	3.5	1		2

System Name
High Reactor Water Level

Category Statement
Knowledge of the interrelations between HIGH REACTOR WATER LEVEL and the following: (CFR: 41.7 / 45.8)

K/A Statement
Main turbine: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q22 295008 K2.08

Other NRC Data

References Provided	None		
K/A Justification	A High RPV water level (Level 8) condition will result in a Main Turbine trip. This question meets the KA because the examinee has to demonstrate knowledge of the interrelationship (trip logic) between high RPV water level and the Main Turbine to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The answer requires recall of system interlocks (1-I).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 9538.08 Rev. 2 • CPS 3305.01 Rev. 12b		
Training Objective	245000.03 DESCRIBE the function, operation, interlocks, trips, and power supplies of the following MAIN TURBINE (TG) System components. .1 Main Turbine Stop Valves		
Previous NRC Exam Use	None		

23	ID: 2104897	Points: 1.00

The plant was operating at rated thermal power when a transient occurred requiring entry into EOP-6 Primary Containment Control.

The CRS has directed the 'B' RO to start all available containment cooling.

What is the reason for performing this action (based on the most limiting component)?

Maintain...

A.	availability of the Wide Range RPV Water Level Instruments.
B.	availability of the Narrow Range RPV Water Level Instruments.
C.	margin to the High Pressure Core Spray (HPCS) Pump NPSH/Vortex Limits.
D.	margin to the Reactor Core Isolation Cooling (RCIC) Pump NPSH/Vortex Limits.

Answer:	A

Answer Justification / Plausibility Statements
A is correct: EOP-6 Primary Containment Control, Containment Temperature Leg, directs holding containment temperature below 122°F, and states that containment temperature affects RPV water level indication. It further directs all available containment cooling to be started if containment temperature cannot be held below 122°F. Figure A RPV Water Level Instruments (4411.07) Detail C Minimum Usable Levels, shows that the Wide Range and Fuel Zone Level Instruments are most affected by rising containment temperature, and that if containment temperature rises above 100°F, the minimum usable WR level indication is -149 inches. In addition, the EOP-TB (page 11-12), states that Wide Range indicated levels increase as containment temperature rises, decrease as drywell temperature rises, and are most sensitive to containment temperature. The Wide Range levels are therefore expressed in terms of containment temperature assuming a drywell temperature of 100°F (the most limiting for the Wide Range instrument since lower drywell temperatures produce higher indicated levels). Incorrect Responses: B is incorrect but plausible. The Narrow Range RPV Water Level instruments are affected by rising Containment Temperatures. Per the EOP-TB page 11-12, Figure C of Detail A lists the levels in terms of whichever temperature has the greater effect upon indicated level; the temperature having the smaller effect is assumed to be at its most limiting value. For the narrow range level instruments, Detail C shows that DW temperature has a greater effect on the Narrow Range level instruments. C is incorrect but plausible. Per the EOP-TB, the NPSH limits are affected by containment heatup, but are defined to be the highest suppression pool temperature which provides adequate net positive suction head for pumps taking suction from the suppression pool. The HPCS Pump is normally aligned to the RCIC Storage Tank, so Detail Z does not apply. If the HPCS suction does transfer to the suppression pool, the NPSH / Vortex Limits are met with Suppression Pool Level > 11 ft. D is incorrect but plausible. Per the EOP-TB, the NPSH limits are affected by containment heatup, but are defined to be the highest suppression pool temperature which provides adequate net positive suction head for pumps taking suction from the suppression pool. The RCIC Pump is normally aligned to the RCIC Storage Tank, so Detail Z does not apply. If the RCIC suction does transfer to the suppression pool, the NPSH / Vortex Limits are as follows: • Suppression Pool Level - 11 ft. • Suppression Pool Temperature - 197°F • Maximum RCIC flow - 700 gpm

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295011.AK3.01	AK3.01	3.6	3.9	1		5

System Name
High Containment Temperature (Mark III Containment Only)

Category Statement
Knowledge of the reasons for the following responses as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) : (CFR: 41.5 / 45.6)

K/A Statement
Increased containment cooling: Mark-III

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q23 295011 K3.01

LP87558.01.03d	LP87558.01.03 Given an EOP-6 condition, discuss the consequences/impact of the following: .01 Failing to reset the CNMT Spray 10 Minute Time Delay Timer on a false initiation, or failure of the initiation signal to clear .02 Operating Standby Gas Treatment at high concentrations of H2 (> 6%) .03 Rising Containment and/or Drywell temperature on RPV water level instrumentation .04 Lowering RPV pressures with elevated Containment and/or Drywell temperatures on RPV water level indication .05 RPV water level indications falling below their minimum usable levels .06 Using VG to evacuate the Containment when Containment temperature is >212° F

	RO Q23 295011 K3.01

LP87558.01.03D	N-CL-OPS-DB-LP87558.01.03 Given an EOP-6 condition, discuss the consequences/impact of the following: .01 Failing to reset the CNMT Spray 10 Minute Time Delay Timer on a false initiation, or failure of the initiation signal to clear .02 Operating Standby Gas Treatment at high concentrations of H2 (> 6%) .03 Rising Containment and/or Drywell temperature on RPV water level instrumentation .04 Lowering RPV pressures with elevated Containment and/or Drywell temperatures on RPV water level indication .05 RPV water level indications falling below their minimum usable levels .06 Using VG to evacuate the Containment when Containment temperature is >212° F

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee is required to choose the reason for starting all available containment cooling in EOP-6 to answer the question.		
SRO-Only Justification	N/A		
Additional Information	Cog Level Justification - this question is a low cog question written at the memory level. The examinee has to recall the bases for performing an action in a procedure (1-F).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A15023)	**Difficulty**	N/A
Technical Reference and Revision #	• EOP-TB Rev. 7 • CPS 4402.01 Rev. 30		
Training Objective	LP87558.01.03 Given an EOP-6 condition, discuss the consequences / impact of the following: .03 Rising Containment and/or Drywell temperature on RPV water level instrumentation		
Previous NRC Exam Use	None		

24	ID: 2104899	Points: 1.00

The plant was operating at rated thermal power when a transient occurred requiring a manual reactor scram.

- Several control rods failed to insert.
- Reactor power is on Range 5 of the IRMs and decaying.
- An insert block is present on the Rod Pattern Control System.

The CRS has directed you to insert control rods to achieve shutdown criteria.

Which of the following actions are permitted to insert control rods?

1) Bypass control rod positions at the Rod Action Control Cabinets (RACS).

2) Raise Drive Water Differential Pressure > 500 psid.

3) Defeat Rod Pattern Controller per CPS 4410.00C012 Defeating ATWS Interlocks.

A.	1 AND 2 ONLY
B.	2 AND 3 ONLY
C.	1 AND 3 ONLY
D.	1, 2 AND 3

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per the power leg of CPS 4404.01 EOP-1A ATWS RPV Control, if reactor power is less than range 7 of the IRMs and lowering and no boron has been injected, then the power leg of EOP-1A is exited and CPS 4100.01 Reactor Scram procedure is entered. CPS 4100.01 Reactor Scram section 4.5 Power Control Actions directs performance of Alternate Rod Insertion per 4411.08 with the exception that the ATWS Interlocks listed in CPS 4410.00C012 shall not be defeated. CPS 4411.08 Alternate Control Rod Insertion Section 2.5 Manual Control Rod Insertion directs /permits the actions in items 1 and 2, but prohibits defeating the Rod Pattern Controller, RPS and ARI logic per CPS 4410.00C012 during low power ATWS conditions. Incorrect Responses: B is incorrect but plausible. • Action 2 is permitted. • Action 3 would be permitted if this were a higher power ATWS and performance of CPS 4410.00C012 Defeating ATWS Interlocks was authorized. C is incorrect but plausible: • Action 1 is permitted. • Action 3 would be permitted if this were a higher power ATWS and performance of CPS 4410.00C012 Defeating ATWS Interlocks was authorized. D is incorrect but plausible: • Actions 1 and 2 are permitted. • Action 3 would be permitted if this were a higher power ATWS and performance of CPS 4410.00C012 Defeating ATWS Interlocks was authorized.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295015.AA1.04	AA1.04	3.4	3.7	1		1

System Name
Incomplete SCRAM

Category Statement
Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM : (CFR: 41.7 / 45.6)

K/A Statement
Rod control and information system: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q24 295015 A1.04

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to operate RC&IS during a low power ATWS to answer the scram.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to analyze the conditions in the stem and then determine required actions based on the analysis (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4404.01 Rev. 30 • CPS 4100.01 Rev. 23f • CPS 4411.08 Rev. 6b		
Training Objective	N-CL-OPS-DB-LP87553.01.05 Given CPS No. 4411.08, ALTERNATE CONTROL ROD INSERTION .03 State the methods that are available for Alternate Control Rod Insertion		
Previous NRC Exam Use	None		

25	ID: 2107088	Points: 1.00

At 0100 the plant was operating at 20% power with the Reactor Mode Switch in RUN.

At 0110, an inadvertent isolation signal caused the following valves to close:

- 1B21-F022A Main Steam Line A Inbd Vlv
- 1B21-F022C Main Steam Line C Inbd Vlv
- 1B21-F028B Main Steam Line B Outbd MSIV

At 0111, APRMs will indicate...

A.	0%.
B.	greater than 0% but less than 20%.
C.	20%.
D.	greater than 20%.

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per 5005.03 (3C) Div 2 or 3 MSIV CI Trip, closure of a MSIV on MSL B or C will result in a reactor scram if any two instrument channels in different steam lines trip on MSIV closure with the Mode Switch in RUN. Incorrect Responses: B is incorrect but plausible. Core thermal power will indicate above 0% for an extended period of time, however APRMs will indicate 0% within one minute. C is incorrect but plausible because a single MSL can pass 20% steam flow, however RPS initiates a scram when MSIVs in 2 or more MSLs are not full open. D is incorrect but plausible because a single MSL can pass 20% steam flow, and closure of multiple MSIVs would initially result in a rise in Reactor pressure which will collapse voids in the reactor and cause power to rise. RPS however, initiates a scram when MSIVs in 2 or more MSLs are not full open so APRMs will indicate 0% 1 minute after the MSIVs go closed.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295020.AA2.03	AA2.03	3.7	3.7	1		5

System Name
Inadvertent Containment Isolation

Category Statement
Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION : (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
Reactor power

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q25 295020 A2.03

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to determine the impact on reactor power during an inadvertent MSIV closure to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the conditions in the stem, and then predict the impact on reactor power to answer the question (2-RI).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 5005.03 Rev. 31c		
Training Objective	212000.09 DISCUSS the effect: .2 A total loss or malfunction of various plant systems has on the Reactor Protection System (RPS) and Alternate Rod Insertion (ARI) System.		
Previous NRC Exam Use	None		

26	ID: 2105004	Points: 1.00

A reactor startup is in progress IAW CPS 3002.01 Heatup and Pressurization.

Conditions are as follows:

- Reactor Power - 6%
- Reactor Pressure - 925 psig

THEN, the operating Control Rod Drive (RD) pump tripped.

Actions are in progress to start the standby RD pump.

If RD cannot be restored, the reactor is first required to be scrammed _____(1)_____.

While performing the scram immediate actions, the following sequence of events occurred:

1. 'A' RO places the Reactor Mode Switch in shutdown. No control rod movement occurred.
2. 'A' RO arms and depresses Manual Scram Pushbuttons and initiates Alternate Rod Insertion (ARI). All control rods fully insert except for 3 control rods at position 02.

EOP-1A ATWS RPV Control _____(2)_____ required to be entered.

A.	(1) immediately after the first control rod scram accumulator for a withdrawn control rod becomes inoperable (2) is

B.	(1) immediately after the first control rod scram accumulator for a withdrawn control rod becomes inoperable (2) is NOT

C.	(1) 20 minutes after the second control rod scram accumulator for a withdrawn control rod becomes inoperable (2) is

D.	(1) 20 minutes after the second control rod scram accumulator for a withdrawn control rod becomes inoperable (2) is NOT

Answer:	C

Answer Justification / Plausibility Statements

C is correct:
 	
Part 1
Per ITS 3.1.5 Control Rod Scram Accumulators:
• Condition B is entered when two or more control rod scram accumulators are inoperable with reactor steam dome pressure ≥ 600 psig.
  • Required action B.1 requires that charging water header pressure to be restored to ≥ 1520 psig within 20 minutes from discovery of Condition B concurrent with charging water header pressure < 1520 psig.
• Condition D is entered when required action and associated completion time of required action B.1 or C.1 not met.
  • Required action D.1 requires the reactor mode switch to be placed in shutdown position immediately.
Part 2
Per EOP-1 RPV Control, EOP-1 will be entered if Reactor Power is above 5% or unknown when scram is required. Since no control rod movement occurred after the Reactor Mode Switch was taken to shutdown, EOP-1 is required to be entered.
If shutdown criteria in CPS 4100.01 Reactor Scram is not met after the mode switch is in shutdown, then EOP-1A ATWS RPV Control is entered and EOP-1 is exited.
The reactor power leg of EOP-1A directs manual scram and ARI to be initiated. With 3 control rods at position 02, shutdown criteria of CPS 4100.01 is met, EOP-1A is exited, and EOP-1 RPV Control is entered.
 	
Incorrect Responses:
 	
A is incorrect but plausible. Part 2 is correct. Part 1 would be correct if RPV pressure was < 600 psig.
 	
B is incorrect but plausible.
 	
• Part 1 would be correct if RPV pressure was < 600 psig.
• Part 2 would be correct if shutdown criteria were met after placing the reactor mode switch in shutdown.
 	
D is incorrect but plausible. Part 1 is correct. Part 2 would be correct if shutdown criteria were met after placing the reactor mode switch in shutdown.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295022	B2.4.08	3.8	4.5	1	2	N/A

System Name
Loss of CRD Pumps

Category Statement
Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q26 295022 2.4.8

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to analyze conditions in the stem and determine how EOP-1A and CPS 4100.01 are used when conditions require a scram following a loss of CRD Pumps.
SRO-Only Justification	N/A
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze conditions in the stem and then determine required actions based on the analysis (3-SPK).

NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• ITS 3.1.5 (3.1-16 and 3.1-17) Amendment No. 95 / 192 • CPS 4401.01 Rev. 30 • CPS 4404.01 Rev. 30 • CPS 4100.01 Rev. 23f		
Training Objective	201001.13 Given Control Rod Drive Hydraulic System key parameter indications and plant conditions, DETERMINE if the [Control Rod Drive Hydraulic] Tech Spec LCOs have been met for one hour or less LCO's. N-CL-OPS-DB-LP87553.01.03 Given a diagram of EOP-1A: .02 State the bases for each individual step/action of EOP-1A		
Previous NRC Exam Use	None		

27	ID: 2107126	Points: 1.00

The plant was in a refueling outage when the fuel handling team reported lowering level in the spent fuel pool.

The following alarms were received in the MCR:

- 5040-4F HIGH FLOW SPENT FUEL STOR POOL LEAK DET
- 5040-2F LOW LEVEL SPENT FUEL STOR POOL
- 5040-3F LOW-LOW SPENT FUEL STOR POOL
- 1RIX-AR016 SPENT FUEL STORAGE ARM high alarm with rising trends.
- 1RIX-PR006A, B, C, AND D FUEL BUILDING EXHAUST PRM high alarm with rising trends.

RP reports:

- rising radiation levels throughout the Fuel Building, and
- normal radiation levels in areas outside the Fuel Building

The leak _____(1)_____ be isolated from the MCR.

To minimize personnel radiation exposure, makeup to the spent fuel pool should be initiated by _____(2)_____.

A.	(1) can (2) opening 1FC038 Surge Tanks Makeup Control Valve
B.	(1) can (2) aligning Cycled Condensate (CY) to the FC Cnmt Rtn Header
C.	(1) can NOT (2) opening 1FC038 Surge Tanks Makeup Control Valve
D.	(1) can NOT (2) aligning Cycled Condensate (CY) to the FC Cnmt Rtn Header

Answer:	D

Answer Justification / Plausibility Statements

D is correct:

Part 1

The alarm procedures listed below direct the following actions:

Alarm Procedure	Off-Normal Procedure	Action
CPS 5040.04 (4F) High Flow Spent Fuel Stor Pool Leak Det	CPS 4011.02 Spent Fuel Pool Abnormal Level Decrease.	Isolate FC system leaks using Table 1 for isolating Fuel Bldg fuel pool liner leakoff flow detection lines. The 4 valves listed in 4011.02 Table 1 (1FC127E, 1FC127F, 1FC127G, and 1FC127H) are located in FB 712.
CPS 5140.63 AR/PR Annunciator - Fuel Building Exhaust 1RIX-PR006A, B, C, D	CPS 4979.01 Abnormal Release of Airborne Radioactivity.	Take appropriate actions as necessary to isolate, reduce or terminate the cause and consequences of the airborne radioactivity.
5140.08 Spent Fuel Storage - 1RIX-AR016	CPS 4979.02 Abnormal High Area Radiation Levels	Take appropriate actions as necessary to isolate, reduce or terminate the cause and consequences of the abnormal area radioactivity.

The Fuel Pool Liner Leakoff Flow Detection Root valves can NOT be closed from the MCR.

Part 2

CPS 4011.02 step 4.3 directs restoring spent fuel pool level/cooling per one of the following methods:

- Preferred: CPS 3317.01 Fuel Pool Cooling (FC) Low Level Spent Fuel Storage Pool Section. CPS 3317.01 step 8.2.1.4 directs adding water to the FC Surge Tanks using 1FC038 FC Surge Tanks Makeup Control Valve from its local control switch. The local control switch for 1FC038 is located on FB 755 and would expose the operator to rising rad levels.
- Alternate: Using CY flush connection to the FC Cnmt Rtn header. This method is aligned using valves that are located and operated in the Turbine Building and the Containment. Since the question requests the makeup method that minimizes personnel exposure, the alternate flow path should be used.

Incorrect Responses:

A is incorrect but plausible:

- Part 1 - A number of FC system leaks can be isolated from the MCR. Specifically, CPS 4011.02 step 4.2 directs isolating FC system leaks using CPS 3317.01 section 8.1.4.12 Manually Isolating Containment Pools. CPS 3317.01 Section 8.1.4.12 directs closing

1FC007, 1FC008, 1FC037, and 1FC036. These 4 valves are motor operated valves operated from MCR Panel 1H13-P800. • Part 2 - Use of 1FC038 is listed as the preferred method listed in CPS 4011.02 section 4.3 for restoring spent fuel pool level, however 1FC038 is operated from FB 755 and would expose the operator to rising radiation levels. B is incorrect but plausible. A number of FC system leaks can be isolated from the MCR. Specifically, CPS 4011.02 step 4.2 directs isolating FC system leaks using CPS 3317.01 section 8.1.4.12 Manually Isolating Containment Pools. CPS 3317.01 Section 8.1.4.12 directs closing 1FC007, 1FC008, 1FC037, and 1FC036. These 4 valves are motor operated valves operated from MCR Panel 1H13-P800. Part 2 is correct. C is incorrect but plausible. Part 1 is correct. Part 2 - Use of 1FC038 is listed as the preferred method listed in CPS 4011.02 section 4.3 for restoring spent fuel pool level, however 1FC038 is operated from FB 755 and would expose the operator to rising radiation levels.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295034.EK1.01	EK1.01	3.8	4.1	1		9

System Name
Secondary Containment Ventilation High Radiation

Category Statement
Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : (CFR: 41.8 to 41.10)

K/A Statement
Personnel protection

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10
41.8	41.8
41.9	41.9

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q27 295034 K1.01

Other NRC Data

References Provided:	None
K/A Justification Statement:	This question meets the KA because the examinee has to determine the operational implications of a Secondary Containment Ventilation High Radiation condition and actions that will provide for personnel protection to answer the question.
SRO Only Justification Statement:	N/A
Additional Information:	Question is written at the analysis and application level (high cog). The examinee has to analyze the conditions in the stem, determine implications and required actions based on that analysis to answer the question (3-SPK).
NRC Exams Only (as applicable)	
Question Type:	New — **Difficulty:** N/A
Technical Reference and Revision #:	• CPS 5040.04 (4F) Rev. 26 • CPS 3317.01 Rev. 33b • CPS 4011.02 Rev. 7c • CPS 5140.63 Rev. 1c • CPS 5140.08 Rev. 0a • CPS 4979.01 Rev. 10e • CPS 4979.02 Rev. 9a
Training Objective:	233000.11 EVALUATE given key Fuel Pool Cooling & Cleanup System parameters, if needed DETERMINE a course of action to correct or mitigate the following abnormal condition(s): .3 Low Level Spent Fuel Pool
Previous NRC Exam Use:	None

28	ID: 2105010	Points: 1.00

The plant was operating at rated thermal power when a 4,000 gpm LOCA and a loss of ALL offsite power occurred.

- ONLY Diesel Generator (DG) 'B' started and loaded to its respective bus.
- Attempts to start the other DGs were unsuccessful.

Plant conditions are as follows:

Parameter	**Value**	**Trend**
Reactor Water Level (WR)	-160	stable
Reactor Water Level (FZ)	-184	trending down
DW Pressure	8.0 psig	trending up
RPV pressure	100 psig	trending down
5065-2A RHR PUMP B AUTO START FAILURE	alarming	N/A
5065-3A RHR PUMP B AUTO TRIP	alarming	N/A
5065-3D RHR B/C INJECTION VALVE PERM TO OPEN annunciator	reset	N/A

Auxiliary Building 737' West is inaccessible.

If the loss of power persists, how is adequate core cooling affected by these conditions?

Adequate core cooling is....

A.	NOT assured.
B.	assured if LPCI Injection Valve control switch for the running pump is placed to OPEN at 1H13-P601.
C.	assured if the LPCI Injection Valve control switch for the running pump is placed to open at the Remote Shutdown Panel.
D.	assured if LPCI Injection Valve control switch for the running pump is placed to OPEN and the Containment Spray Delay Timer Reset pushbutton at 1H13-P601 is depressed every 9 minutes.

Answer:	A

Answer Justification / Plausibility Statements
A is correct: The stem states that the plant experienced a LOCA with a loss of all offsite power. The expected response is: • all 3 Divisional DGs start and power their respective busses • all 3 divisions of ECCS start and inject when their respective injection permissives are met. Since only Div 2 DG started and loaded its respective bus, the only available ECCS systems are LPCI 'B' and 'C'. Annunciators 5065-2A and 5065-3A in alarm indicate that LPCI 'B' failed to auto start, and then tripped when manually started. This leaves LPCI 'C' as the only remaining available injection system. With 5065-3D reset, however, the permissive to open 1E12-F042C LPCI Fm RHR C Shutoff Valve is not met, preventing 1E12-F042C from opening. 1E12-F042C is located in Aux Building 737' at AC-105 (west side). With the west side of AB 737' inaccessible, 1E12-F042C cannot be manually opened. Therefore, with all the above conditions present, adequate core cooling is not assured. Incorrect Responses: B is incorrect but plausible because OP-AA-101-113 Operator Fundamentals Attachment 2 directs operators to take manual actions when automatic actions do not occur. In this case however, 1E12-F042C LPCI Fm RHR C Shutoff Valve will not open with the injection valve permissive to open not met. C is incorrect but plausible because CPS 4003.01C006 RSP - Div 1 LPCI Operation directs operators to align LPCI from the Remote Shutdown Panel. Controls exist only for Div 1 so Div 2 LPCI cannot be initiated from the RSP. D is incorrect but plausible. This answer is partially correct for realigning LPCI 'B' with an automatic containment spray signal present. Incorrect because RHR Pump 'B' does not have power and the conditions for automatic containment spray do not exist.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
203000.K3.04	K3.04	4.6*	4.6*	2		2

System Name
RHR/LPCI: Injection Mode (Plant Specific)

Category Statement
Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following: (CFR: 41.7 / 45.4)

K/A Statement
Adequate core cooling

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

203000.03	203000.03 DESCRIBE the function, operation, interlocks, trips, physical locations, and power supplies of the following RESIDUAL HEAT REMOVAL System components. .1 Suppression Pool Suction Strainer .2 RHR Pumps .3 RHR Heat Exchangers .4 B/C Water Leg Pump .5 Containment Spray Header and Nozzles .6 Suppression Pool Suction Valves F004A, F004B. .7 RH Shutdown Cooling Suction Valves F006A and F006B .8 Shutdown Cooling Inboard and Outboard Isolation Valves F009 and F008 .9 RH B Supply to Reactor Head Spray Valve F023 .10 RH C Full Flow Test Valve F021 .11 RH A (B) Full Flow Test Valves F024A and F024B .12 RH A (B) Containment Outboard Isolation Valves F027A and F027B .13 RH A(B) Containment Spray A(B) Shutoff Valves F028A and F028B .14 RH A(B) to Containment Pool Cooling Shutoff Valves F037A and F037B .15 LPCI From RH Shutoff Valves F042A, F042B, and F042C .16 LPCI From RH Testable Check Valves F041A, F041B, and F041C .17 RH Heat Exchanger Inlet Valves F047A and F047B .18 RH Heat Exchanger Outlet Valves F003A and F003B .19 RH B Radwaste First and Second Isolation Valves F049 and F040 .20 RH Heat Exchanger Bypass Valves F048A and F048B .21 RH to Feedwater Shutdown Cooling Return Valves F053A and F053B .22 RH Heat Exchanger First and Second Sample Valves F060A, F060B, F075A and F075B .23 RH Pump Minimum Flow Recirc Valves F064A, F064B and F064C .24 RH Heat Exchanger SSW Inlet and Outlet Valves F014A, F014B, F068A and F068B .25 RH Fuel Pool Cooling Assist Suction Valve F066

	Q28 203000 K3.04

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of the impact of a loss of LPCI injection on the ability to maintain adequate core cooling to answer the question.
SRO-Only Justification	N/A
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze several parameters in the stem and then determine predicted outcome based on the analysis (3-PEO).
NRC Exams Only	
Question Type	Bank (CL-LC-1397) — **Difficulty** N/A
Technical Reference and Revision #	• CPS 5065.02 (2A) Rev. 30a • CPS 5065.03 (3A & 3D) Rev. 28a
Training Objective	203000.03 DESCRIBE the function, operation, interlocks, trips, physical locations, and power supplies of the following RESIDUAL HEAT REMOVAL System components. .2 RHR Pumps .15 LPCI From RH Shutoff Valves F042A, F042B, and F042C
Previous NRC Exam Use	None

29	ID: 2105012	Points: 1.00

Plant conditions are as follows:

- Reactor coolant temperature is 234°F
- Reactor coolant heatup rate is 7° per minute

It will take _____ minutes to reach the RHR Shutdown Cooling high pressure isolation interlock setpoint.

A.	9 to 11
B.	14 to 16
C.	17 to 19
D.	32 to 34

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per CPS 4001.02C001 Automatic Isolation Checklist, the Group 3 RHR S/D Cooling isolation occurs at 104 psig (Condition X). Per the saturated steam tables, the saturation temperature for 105 psig is 341°F. At a heatup rate of 7° per minute, the high pressure isolation will occur in ~15 minutes. (341 - 234) / 7 = 15.29 minutes. Incorrect Responses: A is incorrect but plausible. 60 psig is the Group 5, 6, 7 isolation setpoint for RCIC. If the examinee uses 60 psig (307°F), it will take ~ 10 minutes to reach 60 psig. (307 - 234) / 7 = 10.43 minutes. C is incorrect but plausible. 125 psig is the RHR 'A' Discharge Pressure Permissive for ADS (5067-4A). If the examinee uses 125 psig (353°F), it will take ~ 17 minutes to reach 125 psig. (353 - 234) / 7 = 17.0 minutes. D is incorrect but plausible. 472 psig is the RHR LPCI Injection Valve Open Permissive (CPS 3312.01 step 8.1.2.4). If the examinee uses 472 psig (465°F), it will take ~ 33 minutes to reach 472 psig. (465 - 234) / 7 = 33.0 minutes.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
205000.K4.02	K4.02	3.7	3.8	2		4

System Name
Shutdown Cooling System (RHR Shutdown Cooling Mode)

Category Statement
Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K/A Statement
High pressure isolation: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q29 205000 K4.02

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of the RHR SDC Cut-In Permissive Pressure / Isolation - High setpoint to answer this question. Specifically, the examinee has to determine the saturation temperature corresponding to the high pressure isolation setpoint, and then determine how long it will take to reach this value given an initial reactor coolant temperature and the heatup rate.		
SRO-Only Justification	N/A		
Additional Information	Cog Level Justification - this is a high cog question written at the application level. The examinee has to calculate the time that it takes to reach a high pressure isolation setpoint given reactor coolant temperature and the heatup rate (3-SPK/SPR).		
NRC Exams Only			
Question Type	Bank (CL-ILT-N14007)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4001.02C001 Rev. 16b • CPS 5067.04 Rev. 31 • CPS 3312.01 Rev. 47		
Training Objective	203000.03 DESCRIBE the function, operation, interlocks, trips, physical locations, and power supplies of the following RESIDUAL HEAT REMOVAL System components. .8 Shutdown Cooling Inboard and Outboard Isolation Valves F009 and F008 .9 RH B Supply to Reactor Head Spray Valve F023 .21 RH to Feedwater Shutdown Cooling Return Valves F053A and F053B		
Previous NRC Exam Use	ILT 14-1 NRC Exam		

30	ID: 2105015	Points: 1.00

The plant is operating at rated thermal power.

THEN, the LPCS Out of Service annunciator (5063-8H) alarmed.

The LPCS Loop Line Break Analog Trip Module (ATM) 1E31-N680A has been found tripped with a differential pressure of -1.4 psid.

LPCS...

A.	CANNOT perform its design function. A break is present in the LPCS injection line outside the shroud.

B.	CANNOT perform its design function. A break is present in the LPCS injection line inside the shroud.

C.	CAN perform its design function. The LPCS injection line is intact. The alarm is due to high core flow at rated conditions.

D.	CAN perform its design function. The break is inside the shroud, however LPCS can provide core cooling by submergence.

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per ITS B3.5.1 ECCS-Operating, the LPCS System consists of a motor driven pump, a spray sparger above the core, piping, and valves to transfer water from the suppression pool to the sparger. The LPCS System is designed to provide cooling to the reactor core when the reactor pressure is low. Per CPS 3315.02 Leak Detection (LD), normal dP between the ECCS lines which penetrate the vessel at operating pressure should be zero. If the HPCS, LPCS, or LPCI line breaks, a pressure transmitter will sense a high dP in one direction. If the dP is over the alarm setpoint, the respective transmitter will actuate an alarm and a "Line Break" Status Light. Per 5063.08 (8H) LPCS OUT OF SERVICE, LPCS Loop line break (1E31-N680A Trip 1: -1.3 psid) as a possible cause for the alarm. Per M10-9075-14, 1E31-N080A monitors differential pressure between the LPCS injection line and the RHR 'A' injection line. The dP cell taps off between the: • LPCS injection line downstream of 1E21-F007 LPCS Manual Shutoff Valve (M05-1073-1 C2 Line 1LP36AA 3/4), and • RHR 'A' injection line between 1E12-F041A LPCI A Inj Testable Chk Vlv and 1E12-F039A LPCI From RHR A Manual Shutoff Valve Therefore, the only way for the dP cell to sense a differential pressure on 1E31-N080A is for the line break to exist outside the core shroud on one of the injection lines (LPCS or RHR 'A'). With 5063-8H locked in and ATM 1E31-N680A indicating -1.4 psid, a LPCS line break outside the core shroud is indicated. Therefore, LPCS cannot perform its design function of spraying the core when RPV pressure is low. Incorrect Responses: B is incorrect but plausible. The first part is correct. The alarm indicates a line break exists, but the leak location is wrong (leak is outside the shroud). C is incorrect but plausible. CPS 3315.02 step 2.2.1 states that the OOS annunciator and status light may not clear until sufficient steaming rate is achieved during startup and may require power and flow to be > 80%. At RTP, the alarm will be reset. D is incorrect but plausible. The line break is outside the shroud. LPCS can inject to the RPV and provide core cooling by submergence, but still cannot fulfill its design function of spraying the core when RPV pressure is low.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
209001.K5.04	K5.04	2.8	2.9	2		2

System Name
Low Pressure Core Spray System

Category Statement
Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : (CFR: 41.5 / 45.3)

K/A Statement
Heat removal (transfer) mechanisms

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q30 209001 K5.04

209001.04	209001.04 STATE the physical location and function of the following LPCS system components, controls, indicators, and/or sensors. .1 LPCS Suction Strainer .2 LPCS Pump, including Controls and Indications .3 LPCS/RHR “A” Water Leg Pump including Controls and Indications .4 LPCS Spray Sparger Including Instrumentation and Indications for Line Break Detection .5 LPCS Pump Suction Valve, including Controls and Indications .6. LPCS PUMP Min Flow Valve including Controls and Indications .7 LPCS Pump Suppression Pool Test Return Valve including Controls and Indications .8 LPCS Injection Valve including Controls and Indications .9 LPCS Testable Check Valve including Controls and Indications .10 LPCS Manual Shutoff Valve including Indications .11 LPCS Pump Room Cooler including Controls and Indications .12 LPCS Pump Suction Pressure Indicator .13 LPCS/RHR “A” Water Leg Pump Suction Pressure Indicator .14 LPCS Out Of Service Switch .15 LPCS MOV Test Prep Switch

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of the operational implication of a line break on the ability of LPCS to remove heat from the core (loss of spray) to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall facts from a procedure to answer the question (1-F).		
NRC Exams Only			
Question Type	Bank (CL-ILT-6443)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3315.02 Rev. 15a • CPS 5063.08 Rev. 35b • ITS B3.5.1 (B3.5-2) Rev. 20-2 • M05-1075-1 Rev. AZ • M05-1073-1 Rev. AH • M10-9075-14 Rev. C		
Training Objective	209001.06 Given a LPCS System Annunciator, DESCRIBE: • The condition causing the annunciator • Any automatic actions • Any operational implications		
Previous NRC Exam Use	None		

31	ID: 2105292	Points: 1.00

The plant is operating at rated thermal power.

4160V Bus 1C1 is powered from the main supply.

THEN, the following events occur:

Time	Event
0200	RAT 'A' locks out on differential overcurrent.
0228	Coolant leak in the Drywell causes DW pressure to reach 1.90 psig.
0232	High jacket water temperature alarm is received for Div 3 DG.

At 0233 the High Pressure Core Spray (HPCS) Pump is...

A.	NOT running.
B.	running with power supplied by DG 1C.
C.	running with power supplied by RAT 'B'.
D.	running with power supplied by the ERAT.

Answer:	D

Answer Justification / Plausibility Statements

D is correct:

Per CPS 5010.01 (1A), the RAT 'A' lockout impacts RAT 'B' because all three RATs are supplied from a single circuit switcher (4538). A fault on any of the three will result in tripping 4538, de-energizing all three transformers. This will result in a lockout trip of the 4160V Bus 1C1 main breaker, and an auto transfer of 1C1 to it's reserve source (ERAT).

Per CPS 5062.02 (2E), high DW pressure (1.68 psig) will result in an auto initiation of the HPCS pump.

Per CPS 3506.01 step 2.1.7, Diesel Generator 1C will trip on overspeed, high generator differential current, low lube oil pressure, high jacket water temperature, reverse power, loss of excitation, overcurrent, and overcrank (this is considered a failure to start, but does cause the lockout relay to trip). All trips except overspeed, generator differential, and overcrank (DG 1A & 1B only), are bypassed on a LOCA signal.

Therefore, at time 0233, the HPCS Pump will be running with power supplied from the ERAT.

Incorrect Responses:

A is incorrect but plausible. This answer would be correct if DW pressure was < 1.68 psig or if the conditions in the stem resulted in deenergizing 4160V Bus 1C1.

B is incorrect but plausible. This answer would be correct if 4160V Bus 1C1 experienced a degraded voltage signal (2nd level UV) instead of a loss of voltage (1st level UV), in which case the bus main and reserve feed breakers would trip and DG 1C would start and synchronize onto the bus.

C is incorrect but plausible. The RAT 'A' lockout impacts RAT 'B' because all three RATs are supplied from a single circuit switcher (4538). A fault on any of the three will result in tripping 4538, deenergizing all three transformers. The answer is plausible because RAT 'A' does not provide power to 4160V Bus 1C1, RAT 'B' does.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
209002.K6.01	K6.01	3.6	3.6	2		2

System Name
High Pressure Core Spray System (HPCS)

Category Statement
Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): (CFR: 41.7 / 45.7)

K/A Statement
Electrical power: BWR-5,6

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q31 209002 K6.01

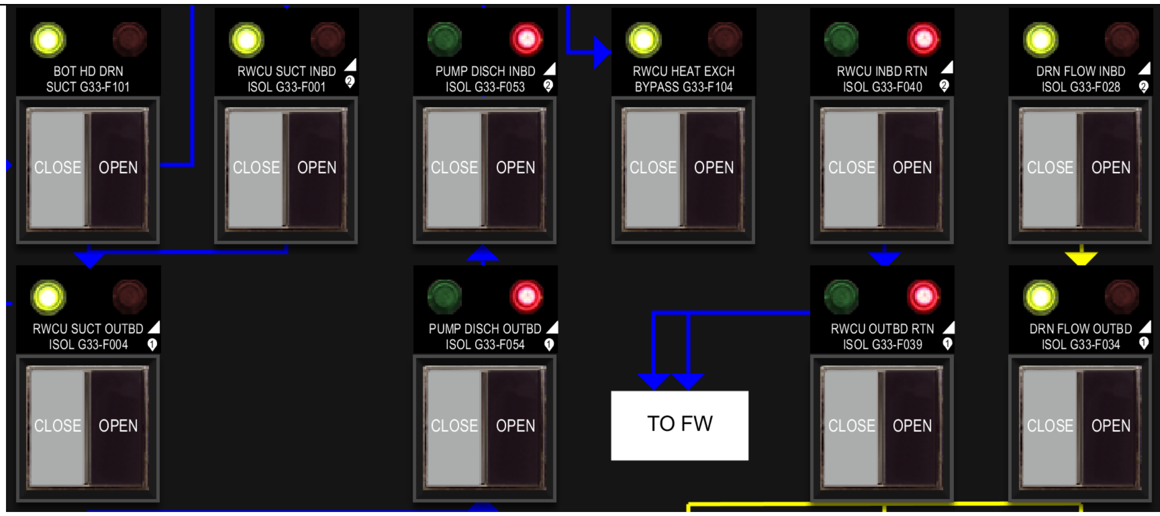
Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of how a loss of AC power will impact the HPCS system to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the series of events in the question stem and then predict an outcome based on the analysis (3-PEO).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A12031)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5010.01 (1A) Rev. 27 • CPS 3506.01 Rev. 39c		
Training Objective	209002.03 DESCRIBE the function, operation, interlocks, trips, physical location, and power supplies of the following HIGH PRESSURE CORE SPRAY System components. .4 HPCS Pump		
Previous NRC Exam Use	None		

32	ID: 2107148	Points: 1.00

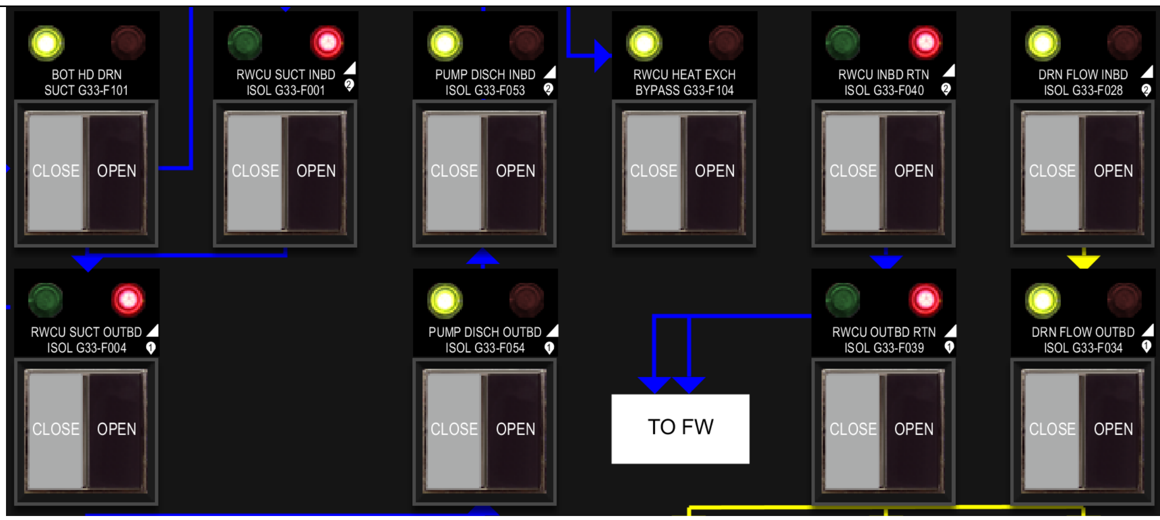
Which of the following graphics shows the expected impact on the Reactor Water Cleanup system valves after the SLC Pumps have been started at 1H13-P601?

1



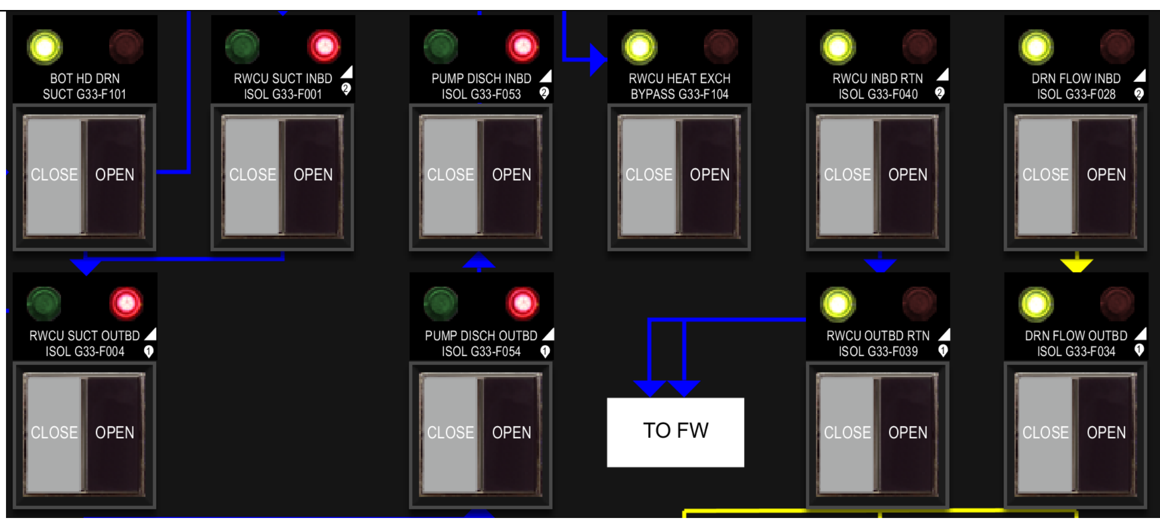
BOT HD DRN SUCT G33-F101
CLOSE OPEN
RWCU SUCT INBD ISOL G33-F001
CLOSE OPEN
PUMP DISCH INBD ISOL G33-F053
CLOSE OPEN
RWCU HEAT EXCH BYPASS G33-F104
CLOSE OPEN
RWCU INBD RTN ISOL G33-F040
CLOSE OPEN
DRN FLOW INBD ISOL G33-F028
CLOSE OPEN
RWCU SUCT OUTBD ISOL G33-F004
CLOSE OPEN
PUMP DISCH OUTBD ISOL G33-F054
CLOSE OPEN
TO FW
RWCU OUTBD RTN ISOL G33-F039
CLOSE OPEN
DRN FLOW OUTBD ISOL G33-F034
CLOSE OPEN

2

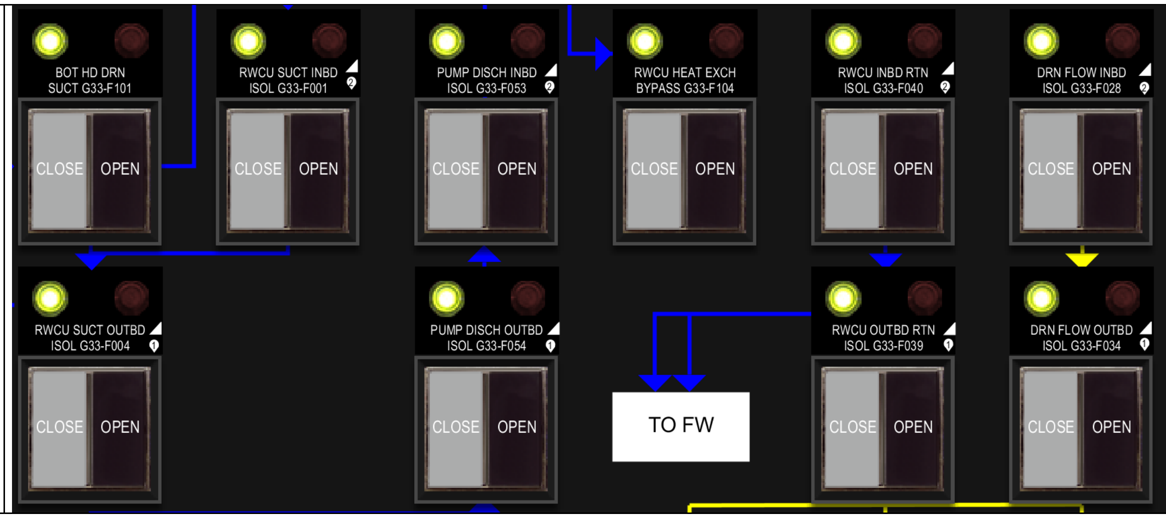


BOT HD DRN SUCT G33-F101
CLOSE OPEN
RWCU SUCT INBD ISOL G33-F001
CLOSE OPEN
PUMP DISCH INBD ISOL G33-F053
CLOSE OPEN
RWCU HEAT EXCH BYPASS G33-F104
CLOSE OPEN
RWCU INBD RTN ISOL G33-F040
CLOSE OPEN
DRN FLOW INBD ISOL G33-F028
CLOSE OPEN
RWCU SUCT OUTBD ISOL G33-F004
CLOSE OPEN
PUMP DISCH OUTBD ISOL G33-F054
CLOSE OPEN
TO FW
RWCU OUTBD RTN ISOL G33-F039
CLOSE OPEN
DRN FLOW OUTBD ISOL G33-F034
CLOSE OPEN

3



BOT HD DRN SUCT G33-F101
CLOSE OPEN
RWCU SUCT INBD ISOL G33-F001
CLOSE OPEN
PUMP DISCH INBD ISOL G33-F053
CLOSE OPEN
RWCU HEAT EXCH BYPASS G33-F104
CLOSE OPEN
RWCU INBD RTN ISOL G33-F040
CLOSE OPEN
DRN FLOW INBD ISOL G33-F028
CLOSE OPEN
RWCU SUCT OUTBD ISOL G33-F004
CLOSE OPEN
PUMP DISCH OUTBD ISOL G33-F054
CLOSE OPEN
TO FW
RWCU OUTBD RTN ISOL G33-F039
CLOSE OPEN
DRN FLOW OUTBD ISOL G33-F034
CLOSE OPEN

4	

BOT HD DRN SUCT G33-F101
CLOSE OPEN
RWCU SUCT INBD ISOL G33-F001
CLOSE OPEN
PUMP DISCH INBD ISOL G33-F053
CLOSE OPEN
RWCU HEAT EXCH BYPASS G33-F104
CLOSE OPEN
RWCU INBD RTN ISOL G33-F040
CLOSE OPEN
DRN FLOW INBD ISOL G33-F028
CLOSE OPEN
RWCU SUCT OUTBD ISOL G33-F004
CLOSE OPEN
PUMP DISCH OUTBD ISOL G33-F054
CLOSE OPEN
TO FW
RWCU OUTBD RTN ISOL G33-F039
CLOSE OPEN
DRN FLOW OUTBD ISOL G33-F034
CLOSE OPEN

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

Answer:	A

Answer Justification / Plausibility Statements
A is correct:

With the RWCU system operating, RWCU system valves are aligned as follows:

Valve Number	Name	Position
1G33-F001	RWCU SUCT INBD ISOL	OPEN
1G33-F004	RWCU SUCT OUTBD ISOL	OPEN
1G33-F053	PUMP DISCH INBD ISOL	OPEN
1G33-F054	PUMP DISCH OUTBD ISOL	OPEN
1G33-F040	RWCU INBD RTN ISOL	OPEN
1G33-F039	RWCU OUTBD RTN ISOL	OPEN

Per CPS 4411.10 step 2.1.2 Verify SLC initiation sequence:

- SLC DISCH TO RPV SQUIB A AND B CONTINUITY lights go out.
- SLC A(B) OUT OF SERVICE annunciators 5067(66)-8F alarm.
- SLC Suct Valve(Vlv) A(B) Fm SLC Strg (Stor) Tank 1C41-F001A(B) valves open.
- **1G33-F001 & F004, RWCU Inbd (Outbd) Suct Isol shut**, unless the isolation logic is bypassed for RPV pressure control.
- SLC Pump A(B), 1C41-C001A(B) start when its respective suction valve is fully open.

This alignment is depicted in Graphic 1.

Incorrect Responses:

B is incorrect but plausible because 1G33-F053 and 54 automatically isolate on a Group 4 isolation signal (CPS 4001.02C001 Automatic Isolation Checklist).

C is incorrect but plausible because 1G33-F040 and 39 automatically isolate on a Group 4 isolation signal (CPS 4001.02C001 Automatic Isolation Checklist).

D is incorrect but plausible because all the RWCU system containment isolation valves will automatically isolate on a Group 4 isolation signal (CPS 4001.02C001 Automatic Isolation Checklist).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
211000.K1.05	K1.05	3.4	3.6	2		1

System Name
Standby Liquid Control System

Category Statement
Knowledge of the physical connections and/or causeeffect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K/A Statement
RWCU

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.2	41.2
41.3	41.3
41.4	41.4
41.5	41.5
41.6	41.6
41.7	41.7
41.8	41.8
41.9	41.9

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

211000.03	211000.03 DESCRIBE the function, operation, interlocks, trips, physical location, and power supplies of the following STANDBY LIQUID CONTROL System components. .1 SLC Storage Tank .2 SLC Storage Tank Heaters .3 SLC Pumps .4 Squib Valves .5 SLC Test Tank .6 Neutron Absorption Solution

	Q32 211000 K1.05

Other NRC Data

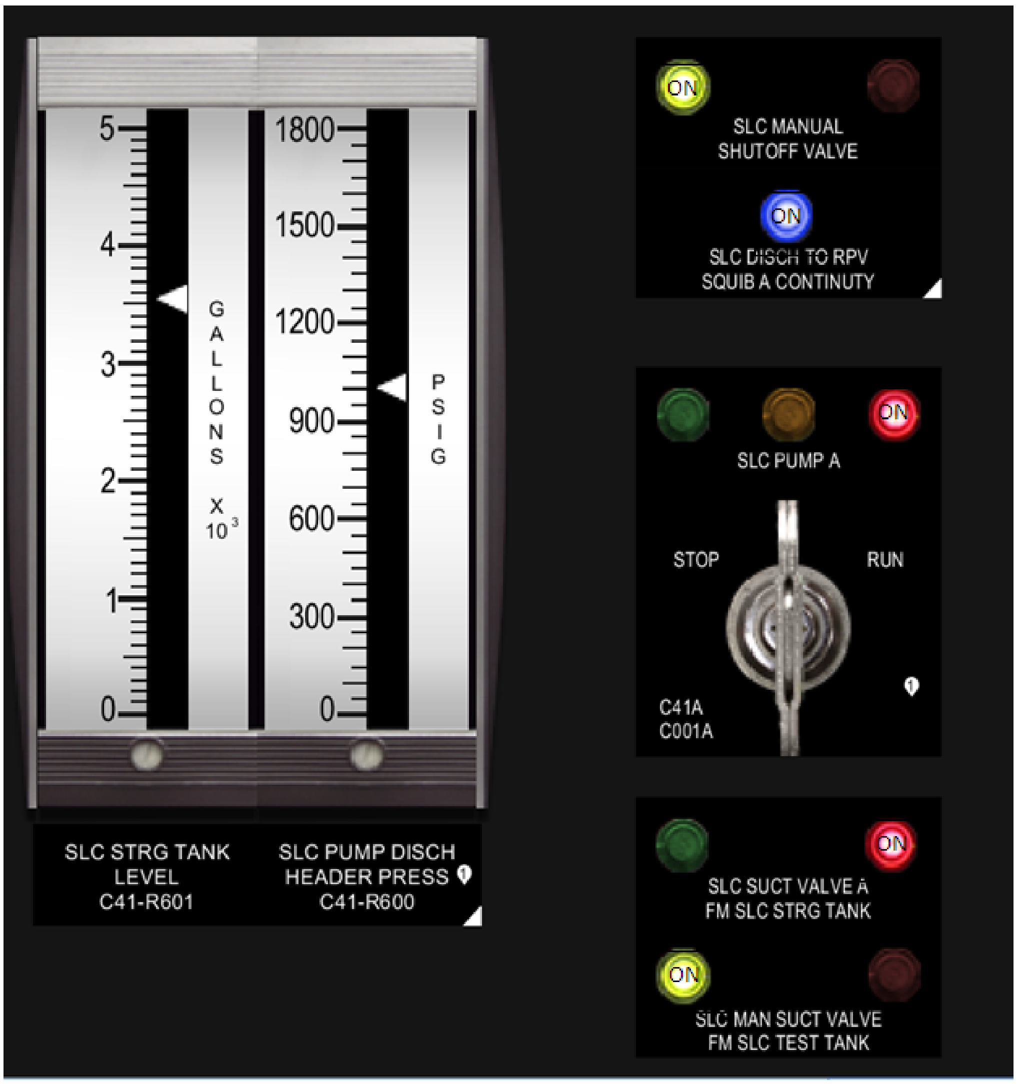
References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of the cause-effect relationship between the SLC system and the RWCU system to answer the question. Specifically, the examinee has to choose a graphic that shows the RWCU system valve alignment after the SLC Pumps have been started.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall the impact of starting the SLC Pumps on the RWCU system and then choose a graphic that depicts that alignment to answer the question (1-I).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A12035)	**Difficulty**	N/A
Technical Reference and Revision #	4411.10 Section 2.1 Rev. 6c		
Training Objective	211000.03 DESCRIBE the function, operation, interlocks, trips, physical location, and power supplies of the following STANDBY LIQUID CONTROL System components. .3 SLC Pumps		
Previous NRC Exam Use	None		

33	ID: 2105325	Points: 1.00

Plant conditions are as follows:

- An ATWS is in progress.
- Reactor pressure is 980 psig.
- SLC has been initiated.

THEN, the following SLC parameters were observed.



Is SLC 'A' injecting into the Reactor? If not, what malfunction(s) exist?

A.	Yes, SLC is injecting into the Reactor.
B.	No; the squib valve has NOT fired.
C.	No; a flowpath to the reactor does NOT exist with the manual shutoff valve closed.
D.	No; the SLC discharge pressure indicates the pump discharge relief valve is lifting.

Answer:	A

Answer Justification / Plausibility Statements

A is correct:

Per CPS 4411.10 SLC Operations step 2.1.4, SLC injection is verified by observing:

SLC Pump Disch Header Press, 1C41-R600 is:

- Slightly > RPV pressure. [If not, SLC is not injecting.]
- Is < 1400 psig. [If > 1400 psig, pump relief may be lifting.]

Following require SLC system response time before verifying.

- Reactor power lowering.
- SLC Strg Tank Level, 1C41-R601 lowering.

Per 5066.06 (6F), the SLC Storage Tank Low Level alarm Level alarms at 3574 gallons which is also the ITS limit on SLC storage tank level. With level in the stem indicated at 3500 gallons, SLC tank level is lower than normal.

Incorrect Distracters:

B is incorrect but plausible. CPS 3314.01 Standby Liquid Control step 2.3 states that when the squib valve is fired, it is possible for the shorted valve assembly to indicate continuity. With pump discharge pressure < 1400 psig and slightly higher than reactor pressure, flow through either the 'A' squib valve and/or the 'B' squib valve is verified.

C is incorrect but plausible. SLC is normally aligned to the HPCS injection line via 1C41-F334. The SLC Manual Shutoff Valve indication on 1H13-P601 indicates the position of 1C41-F008 SLC Injection - Bottom Head which is normally a locked closed valve.

D is incorrect but plausible. This answer would be correct if SLC Pump Discharge Header Pressure was > 1400 psig (SLC Pump Discharge Relief lift setpoint).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
211000.A1.03	A1.03	3.6	3.6	2		1

System Name
Standby Liquid Control System

Category Statement
Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5)

K/A Statement
Pump discharge pressure

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q33 211000 A1.03

211000.16	211000.16 EVALUATE the following STANDBY LIQUID CONTROL indications/responses and DETERMINE if the indication/response is expected and normal. .1 System Initiation .2 Indication of boron injection .3 System Testing

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must be able to determine SLC injection status from the SLC Pump Discharge Pressure indication provided in the stem to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze conditions in the stem and then determine if SLC is injecting to the core based on that analysis (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-N12045)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4411.10 Rev. 6c • CPS 3314.01 Rev. 12a • CPS 3314.01V001 Rev. 10a • CPS 5066.06 (6F) Rev. 25d		
Training Objective	211000.16 EVALUATE the following STANDBY LIQUID CONTROL indications/responses and DETERMINE if the indication/response is expected and normal. .1 System Initiation		
Previous NRC Exam Use	ILT 12-1 NRC Exam		

34	ID: 2105328	Points: 1.00

The plant is operating at rated thermal power.

CPS 9031.16 Manual Scram Channel Functional / SDV Hi Level Bypass Test is in progress.

When the Div 1 Manual Scram pushbutton is armed and depressed, _____(1)_____ of the scram solenoids will de-energize.

An actual single rod scram is prevented by _____(2)_____.

A.	(1) one quarter (2) verifying the A and B scram solenoids > ambient temperature
B.	(1) one half (2) verifying the A and B scram solenoids > ambient temperature
C.	(1) one quarter (2) removing Transient Test scram channels 286 - 289 from Sentinel Trip/Alarm
D.	(1) one half (2) removing Transient Test scram channels 286 - 289 from Sentinel Trip/Alarm

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per CPS 9031.16 Manual Scram Channel Functional / SDV Hi Level Bypass Test, step 5.1 states that an RPS 1/2 scram is generated during manual scram functional tests. In addition, when the Div 1 Manual Scram pushbutton is armed and depressed (steps 8.1.3 and 8.1.5), the Div 1 and 4 'A' solenoids, and the Div 2 and 3 'B' solenoids de-energize. This represents 1/2 of the scram solenoids. To prevent an actual single rod scram, CPS 9031.16 step 8.1.2 requires that solenoids for withdrawn control rods are checked locally > ambient temperature. Incorrect responses: A is incorrect but plausible. The 4 RPS channels are arranged in a 2 out of 4 logic arrangement, with each division supplying 1/4 of the logic inputs. The second part is correct. C is incorrect but plausible: • The 4 RPS channels are arranged in a 2 out of 4 logic arrangement, with each division supplying 1/4 of the logic inputs. • CPS 9031.16 step 5.4 directs operations to remove Transient Test scram channels 286-289 from Sentinal Trip/Alarm to prevent inadvertently actuating the TT system during the surveillance test, not to prevent single rod scrams from occurring. D is incorrect but plausible. CPS 9031.16 step 5.4 directs operations to remove Transient Test scram channels 286-289 from Sentinal Trip/Alarm to prevent inadvertently actuating the TT system during the surveillance test, not to prevent single rod scrams from occurring.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
212000.A2.03	A2.03	3.3	3.5	2		7

System Name
Reactor Protection System

Category Statement
Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

K/A Statement
Surveillance testing

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q34 212000 A2.03

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because it tests the candidates ability to predict how RPS surveillance testing impacts the RPS system, and what procedural actions are required to control the consequences of the surveillance testing.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. It tests recall of facts contained in a procedure (1-F).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 9031.16 Rev. 31c		
Training Objective	212000.05 Discuss the Reactor Protection System (RPS) and Alternate Rod Insertion (ARI) system automatic functions/interlocks including purpose, signals, set points, sensing points, when bypassed, how/when they are. .1 Manual Scram Switches		
Previous NRC Exam Use	None		

35	ID: 2105327	Points: 1.00

A reactor startup is in progress with power on Range 6 of the Intermediate Range Monitors (IRMs).

THEN, annunciator 5005-2H RPS CH B IRM UPSC TRIP OR INOP was received due to a trip on IRM 'B'.

If IRM _____(1)_____ trips, the reactor will scram.

The IRM Upscale Trip or Inop scram is bypassed when the _____(2)_____.

A.	(1) 'F' (2) Reactor Mode Switch is in RUN ONLY
B.	(1) 'G' (2) Reactor Mode Switch is in RUN ONLY
C.	(1) 'F' (2) Reactor Mode Switch is in RUN OR IRM 'B' is on Range 1
D.	(1) 'G' (2) Reactor Mode Switch is in RUN OR IRM 'B' is on Range 1

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per CPS 3306.01 Source/Intermediate Range Monitors (SRM/IRM): Step 2.1 states that IRM Channel 'B' & 'F' comprise the 2 IRMs in Division 2. IRM 'C' & 'G' comprise the 2 IRMs in Division 3. Step 2.5 states that the IRM system inputs the IRM channel Upscale Trip (120/125) and Inoperative Trip into the RPS system. To cause a scram, one of the two IRM channels in two of the four divisions must develop one of these trips. RPS trips from the IRMs are bypassed when the mode switch is in RUN. Therefore, with IRM 'B' tripped (Div 2), an IRM in a different division must trip to get an RPS trip. IRM 'G' (Division 3) meets this criterion. Incorrect Responses: A is incorrect but plausible. IRM 'F' provides a trip signal to RPS, however, a trip of IRM 'F' (Div 2) and IRM 'B' does not satisfy the 2 out of 4 RPS logic requirement. C is incorrect but plausible: • IRM 'F' provides a trip signal to RPS, however, a trip of IRM F (Div 2) does not satisfy the 2 out of 4 RPS logic requirement. • IRM downscale rod blocks are bypassed with the associated IRM channel on range 1, but RPS trips from the IRMs are only bypassed with the RMS in Run. D is incorrect but plausible. IRM downscale rod blocks are bypassed with the associated IRM channel on range 1, but RPS trips from the IRMs are only bypassed with the RMS in Run.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
215003.A3.02	A3.02	3.3	3.3	2		7

System Name
Intermediate Range Monitor (IRM) System

Category Statement
Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: (CFR: 41.7 / 45.7)

K/A Statement
Annunciator and alarm signals

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

215003.05	215003.05 Discuss the IRM system automatic functions/interlocks including purpose, signals, set points, sensing points, when bypassed, how/when they are. .1 Scrams .2 Control Rod Withdrawal Blocks

	Q35 215003 A3.02

Other NRC Data

References Provided	None		
K/A Justification	KA justification - this question meets the KA because the examinee must be able to determine if a scram would be initiated given the alarm indications provided in the answer choices.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall interlocks associated with the IRM system to answer the question (1-I).		
NRC Exams Only			
Question Type	Bank (CL-ILT-N12052)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5005.02 (2H) Rev. 29e • CPS 3306.01 Rev. 12b		
Training Objective	215003.05 Discuss the IRM system automatic functions/interlocks including purpose, signals, set points, sensing points, when bypassed, how/when they are. .1 Scrams		
Previous NRC Exam Use	12-1 NRC Exam		

36	ID: 2106702	Points: 1.00

A reactor startup is in progress.

Criticality was achieved at 08:17.

The following data was recorded into Table 3 Manual Period (Doubling Time) Calculation of CPS 3001.01 Preparation For Startup & Approach To Critical:

Table 3: Manual Period (Doubling Time) Calculation (OK to duplicate this Table and attach to procedure as needed.)						
Date/Time	SRM/IRM Channel	SRM/IRM Reading	SRM/IRM Reading Doubled	Elapsed Time for SRM/IRM Reading to Double (Seconds)	Period =	Initials
06/15/13 08:17:00	A	600	1200	220		

No additional control rod motion has occurred.

Doubling times for the remaining SRM channels (B, C, and D) are consistent with SRM 'A'.

Actual period is between _____(1)_____ seconds.

At the current period, additional Shift Manager approval to continue rod withdrawals _____(2)_____ required.

A.	(1) 147 - 157 (2) is

B.	(1) 308 - 330 (2) is

C.	(1) 147 - 157 (2) is NOT

D.	(1) 308 - 330 (2) is NOT

Answer:	D

Answer Justification / Plausibility Statements

D is correct:

Part 1

Per CPS 3001.01 Preparation For Startup & Approach To Criticality, Note before step 8.2.3 and Table 3: Manual Period (Doubling Time) Calculation, period may be calculated by multiplying the amount of time it takes for indicated power level to double by a factor of 1.443.

- **220 * 1.443 = 317.46**
- The range was derived by multiplying 220 by 1.4 and 1.5 (308 - 330)

Part 2

CPS 3001.01 step 4.11 states that all rod pulls between criticality and the point of adding heat need to be carefully analyzed for impact on reactor period. When the reactor period is below 300 seconds, extra caution should be taken to prevent a short period condition. By restricting rod withdrawals when reactor period is less than 300 seconds, reactivity margin is maintained to account for uncertainties in rod worth predictions that could lead to a short period condition.

Do not withdraw control rods between criticality and the point of adding heat when reactor period is less than 300 seconds unless:

- The predicted rod withdrawal(s) will not bring the reactor to less than a 100 second period, and
- **the Shift Manager approves of the rod withdrawal(s).**

Incorrect Responses:

A is incorrect but plausible. Part 1 is derived by **dividing** the doubling time by 1.4 and 1.5 instead of multiplying. For part 2, if reactor period were less than 300 seconds, additional Shift Manager approval is required to continue rod withdrawals.

B is incorrect but plausible. Part 1 is correct. For part 2, if reactor period were less than 300 seconds, additional Shift Manager approval is required to continue rod withdrawals.

C is incorrect but plausible. This answer is derived by **dividing** the doubling time by 1.4 and 1.5 instead of multiplying. Part 2 is also incorrect but plausible because older revisions of 3001.01 listed the desired period at 100 - 150 seconds before the new guidance was implemented.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
215004.A4.01	A4.01	3.9	3.8	2		7

System Name
Source Range Monitor (SRM) System

Category Statement
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

K/A Statement
SRM count rate and period

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

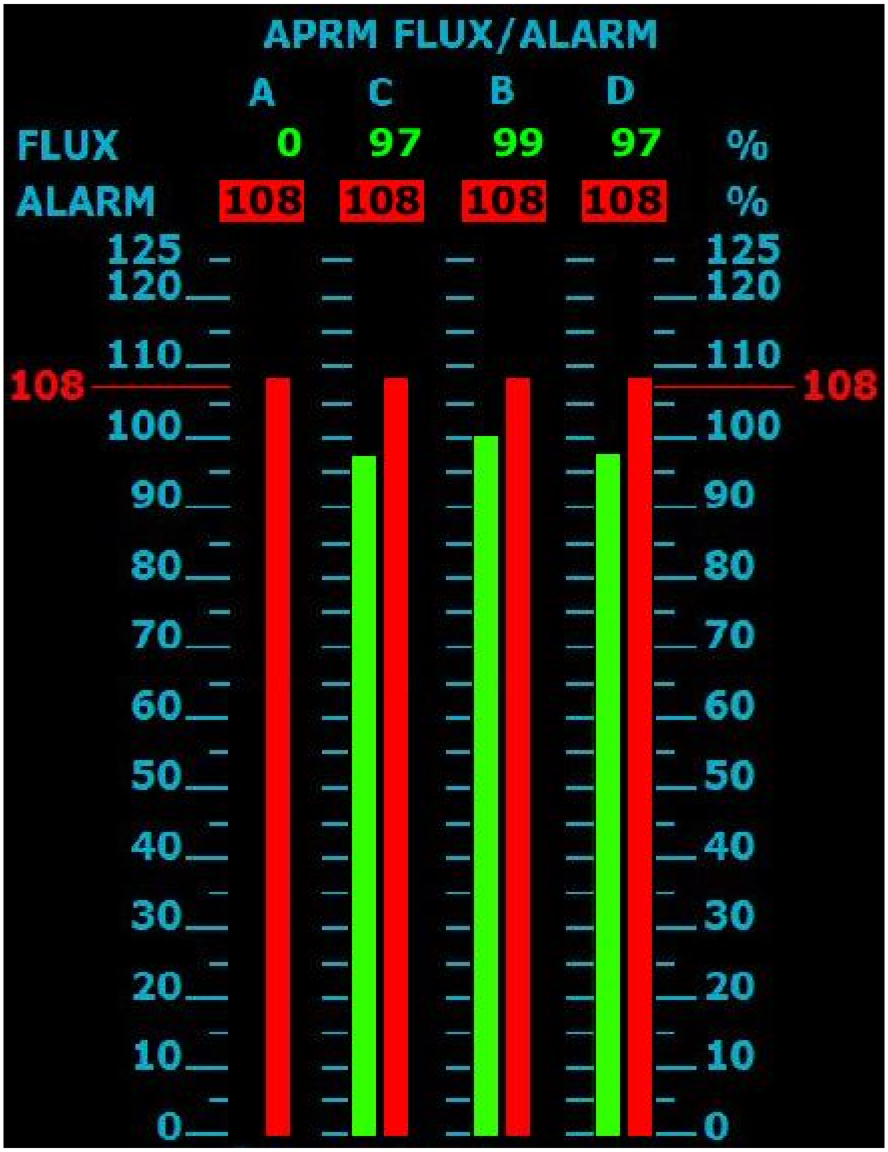
	Q36 215004 A4.01

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to monitor SRM count rate and calculate period to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to solve a problem using knowledge to answer the question (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT- N12048)	**Difficulty**	N/A
Technical Reference and Revision #	CPS 3001.01 Rev. 28e		
Training Objective	RXTH-K3.09 Define doubling time and calculate it using the power equation. LP87435.01.04 Recall the following CPS 3001.01 limitations while operating the plant or on an exam: .4 When control rod manipulations		
Previous NRC Exam Use	ILT 12-1 NRC Exam		

37	ID: 2105318	Points: 1.00

The plant was operating at rated thermal power when the following Plant Process Computer (PPC) indications were observed:



Which of the following describes the expected consequences (if any)?

A.	Scram
B.	NO scram, NO rod block
C.	Rod withdrawal and insertion is blocked.
D.	Rod withdrawal is blocked, rod insertion is permitted.

Answer:	D

Answer Justification / Plausibility Statements
D is correct: The PPC indications show that APRM 'A' has failed downscale. Per CPS 5004-1L APRM DNSC, an APRM downscale condition with the Reactor Mode Switch in RUN will result in a rod block. Per CPS 5006.02 (2H) ROD OUT BLOCK, a neutron monitoring system alarm or inoperable signal will cause rod withdrawal to be blocked. Restrictions on rod insertion are imposed by the Rod Pattern Controller, which is <u>not</u> in effect with the reactor at rated thermal power. Incorrect Responses: A is incorrect but plausible because the 4 PPC alarm indications are showing red. The PPC indications coincide with the APRM upscale rod blocks. If the PPC flux indicators were above the APRM Upscale flux trip, an RPS trip would be initiated. B is incorrect but plausible with all APRMs below the high flux rod block and RPS trip setpoints. C is incorrect but plausible because insertion rod blocks can be initiated by the Rod Pattern Controller if a rod selected for movement violates the rod pattern data built into the controller.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.215005	B2.1.19	3.9	3.8	2	1	N/A

System Name
Average Power Range Monitor/Local Power Range Monitor System

Category Statement
Ability to use plant computers to evaluate system or component status. (CFR: 41.10 / 45.12)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q37 215005 2.1.19

215005.16	215005.16 EVALUATE the following AVERAGE POWER RANGE MONITOR indications/responses and DETERMINE if the indication/ response is expected and normal. .1 LPRM failure Downscale .2 LPRM failure Upscale .3 Degraded detector power supply .4 Bypass of a failed LPRM .5 APRM failure Downscale .6 APRM failure Upscale .7 Loss of power to an APRM Channel .8 Loss of Flow Signal

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to use plant computers to evaluate the APRM indications and determine status to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to analyze the indications in the stem and then determine expected response based on the analysis (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-1142371)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5004.01 (1L) Rev. 28c • CPS 3304.02 Rev. 22e • CPS 5006.02 (2H) Rev. 28d		
Training Objective	215005.16 EVALUATE the following AVERAGE POWER RANGE MONITOR indications/responses and DETERMINE if the indication/ response is expected and normal. .5 APRM failure Downscale		
Previous NRC Exam Use	N/A		

38	ID: 2105316	Points: 1.00

The plant is operating at rated thermal power.

Div 1 RCIC Isolation Bypass switch on 1H13-P632 is in BYPASS.

THEN, a valid high RCIC Room temperature condition (greater than the isolation setpoint) is received.

Which of the following RCIC System isolation valves receive an isolation signal?

1	1E51-F076	RHR & RCIC Stm Supp Warm Up Isol Valve
2	1E51-F063	RHR & RCIC Stm Supp Inbd Isol Valve
3	1E51-F064	RHR & RCIC Stm Supp Outbd Isol Valve
4	1E51-F031	RCIC Suppr Pool Suction Valve

A.	1 AND 2 ONLY
B.	1 AND 3 ONLY
C.	3 AND 4 ONLY
D.	2 AND 4 ONLY

Answer:	A

Answer Justification / Plausibility Statements

A is correct:

Per CPS 4001.02C001 Automatic Isolation Checklist, RCIC Room Ambient Temperature - High will result in a Group 5 & 6 isolation of the RCIC system. Groups 5 and 6 consist of the following valves:

Group	Valve Number	Valve Name	Associated Division
5	1E51-F076	RHR & RCIC Stm Supp Warm Up Isol Valve	2
5	1E51-F063	RHR & RCIC Stm Supp Inbd Isol Valve	2
6	1E51-F064	RHR & RCIC Stm Supp Outbd Isol Valve	1
6	1E51-F031	RCIC Suppr Pool Suction Valve	1

Per CPS 9432.13A RCIC Equipment Room Temp. 1E31-N602A Channel Calibration and CPS 9432.13B RCIC Equipment Room Temp. 1E31-N602B Channel Calibration the RCIC isolation logic functions as follows:

- Group 5 provides Division 2 (Inboard) isolation protection for RCIC and Group 6 provides Division 1 (Outboard) isolation protection.
- RCIC Leak Detection Bypass Switches will bypass the area temperature isolation signals from Division 1 or 2 respectively.
- A valid high RCIC Room ambient temperature condition with Div 1 RCIC Leak Detection Bypass switch in BYPASS will result in automatic closure of the Division 2 valves (1E51-F076 and 1E51-F063) only.

Incorrect Responses:

B is incorrect but plausible. 1E51-F064 is a Group 6 isolation valve which receives an isolation signal on RCIC Equipment Room High Temperature, but the isolation is prevented by using the Div 1 RCIC Leak Detection Isolation Bypass Switch.

C is incorrect but plausible. 1E51-F064 and 1E51-F031 are both Group 6 isolation valves. Both valves receive an isolation signal on RCIC Equipment Room High Temperature, but the isolation is prevented by using the Div 1 RCIC Leak Detection Isolation Bypass Switch.

D is incorrect but plausible. 1E51-F063 and 1E51-F031 will isolate on a valid high RCIC room temperature condition, however 1E51-F031 isolation is prevented with the Div 1 RCIC Leak Detection Isolation Bypass Switch in BYPASS.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
217000.K1.07	K1.07	3.1	3.2	2		2

System Name
Reactor Core Isolation Cooling System (RCIC)

Category Statement
Knowledge of the physical connections and/or causeeffect relationships between REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K/A Statement
Leak detection

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.2	41.2
41.3	41.3
41.4	41.4
41.5	41.5
41.6	41.6
41.7	41.7
41.8	41.8
41.9	41.9

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q38 217000 K1.07

217000.03	217000.03 DESCRIBE the function, operation, interlocks, trips, physical location, and power supplies of the following REACTOR CORE ISOLATION COOLING System components. .1 Steam Supply Shutoff Valve .2 Turbine Trip Throttle Valve .3 Exhaust Line Rupture Discs .4 Exhaust Vacuum Breakers .5 Exhaust Vacuum Breaker line Isolation Valves .6 Pump Suction Valves .7 Min Flow Valve .8 Water Leg Pump .9 Gland Seal Air Compressor .10 Lube Oil System .11 RCIC Room Cooling System .12 Ramp Generator

Other NRC Data

References Provided	None		
K/A Justification	KA Justification - This question meets the KA because the examinee must have knowledge of the effect of bypassing RCIC Leak Detection logic (which constitutes a loss of the Group 5 / 6 isolation logic) has on the RCIC system.		
SRO-Only Justification	N/A		
Additional Information	This question meets the definition of a high cog question because the examinee is required to predict the effect of an isolation signal with RCIC Leak Detection Bypass Switch in an abnormal configuration (2-RI).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A12033)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4001.02C001 Rev. 16b • CPS 9432.13A Rev. 1c • CPS 9432.13B Rev. 1b		
Training Objective	223006.03 DESCRIBE the function, operation, interlocks, trips, physical location, and power supplies of the following LEAK DETECTION System components. 4. Area Temperature Monitoring		
Previous NRC Exam Use	None		

39	ID: 2105314	Points: 1.00

A plant event has occurred, resulting in the following conditions:

Parameter	Value	Trend
RPV Water Level	-140" Wide Range	lowering at 1 inch/min
Containment pressure	3.4 psig	rising at 0.1 psig/min
DW pressure	6.2 psig	rising at 0.1 psig/min

Appropriate automatic actions were verified.

THEN, 125V DC MCC 1A de-energized.

If NO operator action is taken and present conditions persist, in ten (10) minutes ADS valves will . . .

A.	be opened by both logic circuits.
B.	be opened by only the Division 2 logic circuit.
C.	**NOT** open due to RPV water level conditions NOT met.
D.	**NOT** open due to containment pressure parameters NOT met.

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per CPS 5066.06 (6A) ADS LOGIC F INITIATED, Div 2 ADS logic will receive an initiation signal if the following conditions exist: • High Drywell pressure <u>and</u> RPV Water Level Low 1 (-145.5") with RHR B or C pump running. Per CPS 4201.01C001 Loss of 125VDC MCC 1A (1DC13E) Load Impact List Ckt. #23 of DC MCC 1A supplies power to the 'A' solenoids of the ADS valves. The valves will still operate with 'B' power (Division 2 logic). Incorrect Responses: A is incorrect but plausible. ADS logic is powered from NSPS Inverters A and B, however the 'A' ADS solenoids have no power. C is incorrect but plausible. During certain transients, ADS is manually initiated at Top of Active Fuel (-160"). With RPV water level in the stem above TAF, it is plausible but incorrect that ADS has not yet reached the RPV level initiation setpoint. D is incorrect but plausible. The containment pressure provided in the stem is below the automatic containment spray actuation setpoint of 22.3 psia (7.6 psig) (5065-3B). Containment pressure does not provide input to the ADS logic.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
218000.K2.01	K2.01	3.1*	3.3*	2		3

System Name
Automatic Depressurization System

Category Statement
Knowledge of electrical power supplies to the following: (CFR: 41.7)

K/A Statement
ADS logic

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q39 218000 K2.01

218000.03	218000.03 DESCRIBE the function, operation, interlocks, trips, physical location and power supplies of the following AUTOMATIC DEPRESSURIZATION (ADS) System components. .1 ADS Logic .2 105 Second Timer .3 6 Minute Timer

CPS.722839	RO COMP #1

218000.07	218000.07 Given the AUTOMATIC DEPRESSURIZATION (ADS) system, DESCRIBE the systems supporting and the nature of the support.

218000.09	218000.09 DISCUSS the effect: .a A total loss or malfunction of the AUTOMATIC DEPRESSURIZATION (ADS) System has on the plant. .b A total loss or malfunction of various plant systems has on the AUTOMATIC DEPRESSURIZATION (ADS) System.

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of how a loss of divisional DC power impacts the ADS logic to answer the question.
SRO-Only Justification	N/A
Additional Information	This question is a high cog question written at the analysis and application level. The examinee has to analyze the parameters in the stem and then determine how ADS is impacted with a loss of DC power (2-RI).

NRC Exams Only			
Question Type	Bank (CL-ILT-435422)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4201.01C001 Rev. 1 • CPS 5065.03 (3B) Rev. 28a • CPS 5066.06 (6A) Rev. 25d • CPS 5067.05 (5A) Rev. 32		
Training Objective	218000.03 DESCRIBE the function, operation, interlocks, trips, physical location and power supplies of the following AUTOMATIC DEPRESSURIZATION (ADS) System components. .1 ADS Logic 218000.07 Given the AUTOMATIC DEPRESSURIZATION (ADS) system, DESCRIBE the systems supporting and the nature of the support. 218000.09 DISCUSS the effect: .b A total loss or malfunction of various plant systems has on the AUTOMATIC DEPRESSURIZATION (ADS) System.		
Previous NRC Exam Use	None		

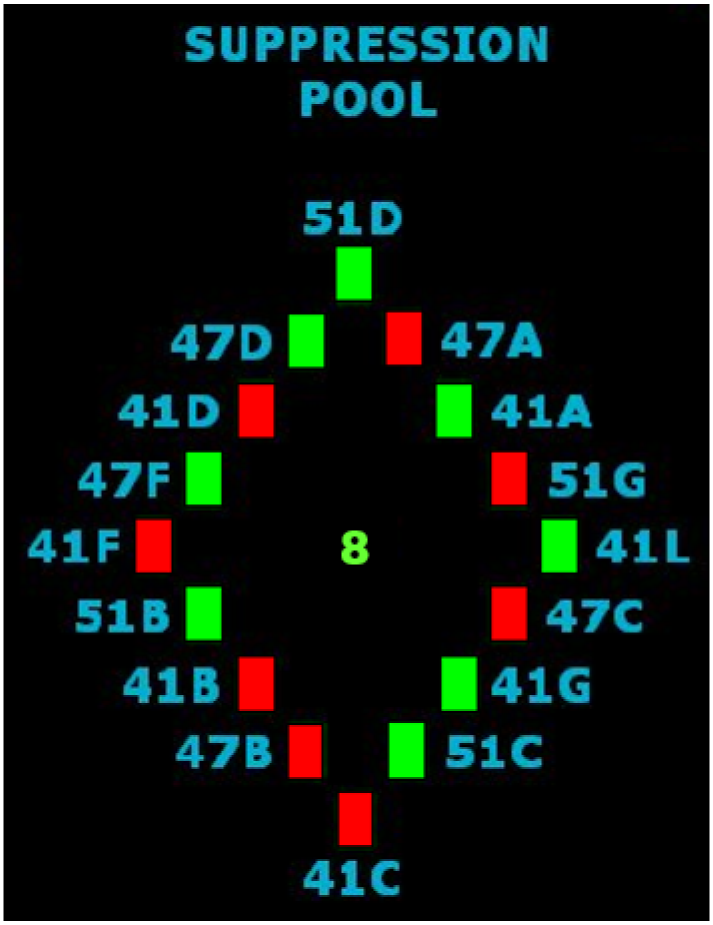
40	ID: 2107204	Points: 1.00

The plant was operating at rated thermal power.

THEN, an automatic initiation of the Automatic Depressurization System (ADS) occurred.

Annunciator _____(1)_____ is actuated by the Safety Relief Valve (SRV) Acoustic Monitors.

(2) Is the Containment Support Display (PPC 2H) shown below indicative of an automatic ADS initiation?



A.	(1) 5067-8L SRV Monitoring System Trouble (2) Yes

B.	(1) 5067-8L SRV Monitoring System Trouble (2) No

C.	(1) 5066-5B ADS or Safety Relief Valve Leaking (2) Yes

D.	(1) 5066-5B ADS or Safety Relief Valve Leaking (2) No

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Part 1 Per CPS 5067.08 (8L), the SRV MONITORING SYSTEM TROUBLE alarm is annunciated by manual opening of any of the ADS SRVs by the SRV Monitoring System. Per CPS 8842.03 Acoustic Monitor System Gain Setting and Verification, step 5.2 states that the SRV Acoustic Monitors provides input to 5067-8L SRV MONITORING SYSTEM TROUBLE. Part 2 Per CPS 3101.01 Main Steam (MS, IS & ADS), step 2.2.3 states that cross talk can be picked up by another SRV's acoustic monitor. This could cause the indication to look as if multiple SRV's are open when there is really only the one open. In the graphic, ADS valves 47A, 51G, 47C, 41C, 41B, 41F, and 41D acoustic monitors are in alarm. SRV 47B (a non-ADS SRV) is showing open due to acoustic monitor cross talk. Incorrect Responses: B is incorrect but plausible. Part 1 is correct. Part 2 is plausible because ADS consists of 7 valves, not 8 as shown. C is incorrect but plausible. 5066-5B ADS OR SAFETY RELIEF VALVE LEAKING is annunciated when any ADS Safety Valve temperature sensor reaches 220°F (CPS 5066.05 (5B)). The second part is correct. D is incorrect but plausible: Part 1 is incorrect because 5066-5B ADS OR SAFETY RELIEF VALVE LEAKING is annunciated when any ADS Safety Valve temperature sensor reaches 220°F, not by its associated acoustic sensor. Part 2 is plausible because ADS consists of 7 valves, not 8 as shown.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
218000.A3.03	A3.03	3.7	3.8	2		3

System Name
Automatic Depressurization System

Category Statement
Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: (CFR: 41.7 / 45.7)

K/A Statement
ADS valve acoustical monitor noise: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q40 218000 A3.03

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to monitor the acoustic monitor indications for the ADS system to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This question is a high cog question written at the analysis and application level. The examinee has to analyze the graphic in the stem and then determine the operational implication on the ADS system to answer the question (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5066.05 (5B) Rev. 28a • CPS 3101.01 Rev. 24 • CPS 8842.03 Rev. 2a • CPS 5067.08 (8L) Rev. 31a		
Training Objective	218000.06 Given a AUTOMATIC DEPRESSURIZATION (ADS) System Annunciator, DESCRIBE: The condition causing the annunciator Any automatic actions Any operational implications		
Previous NRC Exam Use	None		

41	ID: 2106708	Points: 1.00

High Pressure Core Spray (HPCS) Water Level 2 and 8 Analog Trip Module (ATM) 1B21-N673C has failed and will NOT trip.

This ATM failure:

_____(1)_____ prevent 1E22-F023 HPCS Test Valve to Suppr Pool from closing on a valid RPV Level 2 signal.

_____(2)_____ prevent 1E22-F004 HPCS to CNMT Outbd Isln Valve from closing on a valid RPV Level 8 signal.

A.	(1) will (2) will
B.	(1) will (2) will NOT
C.	(1) will NOT (2) will
D.	(1) will NOT (2) will NOT

Answer:	C

Answer Justification / Plausibility Statements

C is correct.

Part 1

Per CPS 9433.07A ECCS Reactor Vessel Water Level B21-N073C Channel Calibration, step 2.1.4 states that ATM B21-N073C provides Reactor Water Level 2 trip signal to the HPCS system, including E22-F023, HPCS TEst Valve to Suppr Pool. A trip of the ATM would cause an isolation of E22-F023 if coincident logic trips are present.

Per ITS B3.3.5.1 ECCS Instrumentation page B 3.3-87, The HPCS System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low, Level 2 or Drywell Pressure-High. The outputs of the ATMs are connected to solid state logic which is arranged in a one-out-of-two taken twice logic for each variable.

The ATMs that provide HPCS initiation signals on RPV Level 2 are the same ATMs that provide input to the CRVICS Group 8 isolation logic for 1E22-F023.

Since the stem states that a single ATM (B21-N673C) has failed, a valid RPV level 2 signal will still cause a CRVICS isolation of 1E22-F023.

Part 2

Per CPS 9433.07A ECCS Reactor Vessel Water Level B21-N073C Channel Calibration, step 2.1.5 states that ATM B21-N073C also provides Reactor Water Level 8 trip signal which would cause closure of E22-F004, HPCS to Cnmt Outbd Isln Valve, if coincident logic trips are present.

Per ITS B3.3.5.1 ECCS Instrumentation page B3.3-88, the HPCS System provides makeup water to the reactor until the reactor vessel water level reaches the high water level (Level 8) trip, at which time the HPCS injection valve closes. The logic is two-out-of-two to provide high reliability of the HPCS system.

Since the stem states that a single ATM (B21-N673C) is failed in the non-tripped condition, a single valid RPV level 8 signal will not initiate a Level 8 isolation of 1E22-F004.

Incorrect responses:

A is incorrect but plausible and is partially correct. Part 1 would be correct if the CRVICS Isolation Logic for 1E22-F023 was configured similarly to the HPCS Level 8 Isolation Logic for 1E22-F004 (two-out-of-two logic). Part 2 is correct.

B is incorrect but plausible. This answer would be correct if:

- the CRVICS Isolation Logic for 1E22-F023 was a two-out-of-two logic configuration, and
- the HPCS Level 8 Isolation Logic was a one-out-of-two taken twice logic configuration.

D is incorrect but plausible and is partially correct. Part 1 is correct. Part 2 would be correct if the HPCS Level 8 Isolation Logic was configured similarly to the CRVICS isolation logic for 1E22-F023 (one-out-of-two taken twice).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
223002.K3.23	K3.23	3.6	3.6	2		5

System Name
Primary Containment Isolation System/Nuclear Steam Supply Shut-Off

Category Statement
Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: (CFR: 41.7 / 45.4)

K/A Statement
High pressure core spray : Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q41 223002 K3.23

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must demonstrate knowledge of the effect that a loss or malfunction the PCIS/NSSS System will have on HPCS to determine the correct response.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall facts to select the correct response (1-F).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 9433.07A Rev. 1b • ITS B3.3.5.1 (B3.3-87 & 88) Revision No. 4-8 • ITS B3.3.5.1 (B3.3-99) Revision No. 7-5		
Training Objective	209002.09 .2 A total loss or malfunction of various plant systems has on the HIGH PRESSURE CORE SPRAY System.		
Previous NRC Exam Use	None		

42	ID: 2105305	Points: 1.00

The plant is operating at rated thermal power.

BOTH hand switches for 1B21-F051G Main Steam Line C ADS Valve/SRV (located on MCR panels H13-P601 and H13-P642) are placed in the OFF position.

Which mode(s) of operation will still cause this SRV to actuate?

A.	ADS mode ONLY.
B.	Relief mode ONLY.
C.	Safety mode AND Relief mode.
D.	Safety mode AND ADS mode.

Answer:	D

Answer Justification / Plausibility Statements
D is correct. Per CPS 3101.01 Main Steam (MS, IS & ADS), 8.2 Abnormal Operations (NOTE) and Annunciator Response Procedure (ARP) for annunciator 5067-5C DIV 1 SFTY/RLF VLV CONT SWITCH IN OFF POSITION: The Pressure relief function of a SRV solenoid will be INOP for any valve with its handswitch in the OFF position. • 'A' solenoid handswitch operation from 1H13-P601 • 'B' solenoid handswitch operation from 1H13-P642 With both handswitches in OFF, the valve will not open in relief mode but will operate in ADS mode (electrical) and as a safety valve (mechanical). Incorrect Responses: A is partially correct (ADS will actuate with the SRV keylock switches in off). Incorrect but plausible because certain operating modes of the SRVs are disabled by taking BOTH handswitches to OFF. In the safety mode, steam pressure acts against a spring loaded disk that will pop open when the valve inlet pressure exceeds the spring force (ITS B3.4.4 pg. B3.4.4-1). B is incorrect but plausible because certain operating modes of the SRVs are NOT disabled by taking BOTH handswitches to OFF. However, relief mode is disabled when both handswitches are taken to off. C is partially correct (Safety Mode will actuate with the SRV keylock switches in off). Incorrect but plausible because certain operating modes of the SRVs are NOT disabled by taking BOTH handswitches to OFF. However, relief mode is disabled when both handswitches are taken to off.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
239002.K4.08	K4.08	3.6	3.7	2		3

System Name
Relief/Safety Valves

Category Statement
Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K/A Statement
Opening of the SRV from either an electrical or mechanical signal .,

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q42 239002 K4.08

239001.15	239001.15 Given MAIN STEAM System initial conditions, PREDICT how the system and/or plant parameters will respond to the manipulation of the following controls.

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must demonstrate knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for opening of the SRV from either an electrical or mechanical signal to determine the correct response.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall facts to select the correct response (1-F).		
NRC Exams Only			
Question Type	Bank (CL-ILT-0741)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3101.01 Rev. 24 • CPS 5067.05 (5C) Rev. 32 • ITS B3.4.4 (B3.4.4-1) Rev. 0		
Training Objective	239001.15 Given MAIN STEAM System initial conditions, PREDICT how the system and/or plant parameters will respond to the manipulation of the following controls.		
Previous NRC Exam Use	None		

43	ID: 2106709	Points: 1.00

Which of the following Safety Relief Valve (SRV) functions are addressed by Technical Specification 3.6 Containment Systems?

A.	Relief

B.	Safety

C.	Low-Low Set (LLS)

D.	Automatic Depressurization System (ADS)

Answer:	C

Answer Justification / Plausibility Statements
C is correct. Per Technical Specification 3.6 Containment Systems and ITS LCO 3.6.1.6 Low-Low Set (LLS) Valves: The LLS function of five safety/relief valves shall be OPERABLE. Incorrect Responses: A and B are incorrect but plausible because the relief mode and the safety mode of the SRVs is required by ITS. However, the relief and safety modes are addressed in ITS LCO 3.4.4 Safety/Relief Valves (S/RVs) as part of the Reactor Coolant System (RCS) specifications. D is incorrect but plausible because the ADS mode is required by ITS. However, the ADS mode is addressed in ITS 3.5 ECCS - Operating. The ADS system is a subsystem of ECCS.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.239002	B2.2.38	3.6	4.5	2	1	N/A

System Name
Relief/Safety Valves

Category Statement
Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
43.1	43.1

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

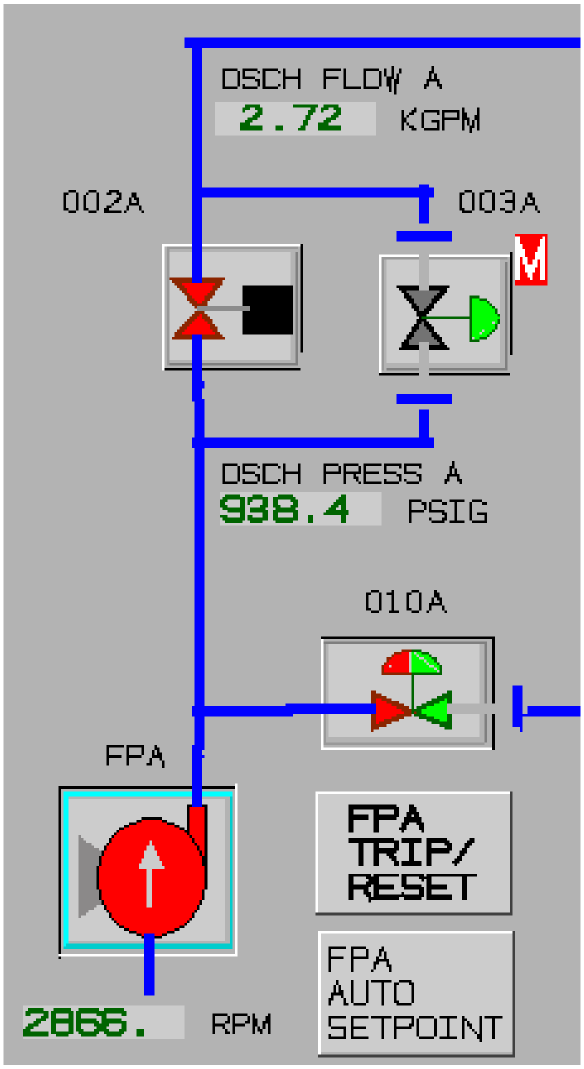
	Q43 239002 2.2.38

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must demonstrate knowledge of conditions and limitations in the facility license to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall facts to select the correct response (1-F).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• ITS LCO 3.4.4 (3.4-10) Amend. 95 • ITS LCO 3.5.1 (3.5-1) Amend. 216 • ITS LCO 3.6.1.6 (3.6-22) Amend. 187		
Training Objective	LP85802.2.2.38 Knowledge of conditions and limitations in the facility license. (Moved from 2.1.10)		
Previous NRC Exam Use	None		

44	ID: 2105289	Points: 1.00

A plant shutdown is in progress with Reactor Power at 24%.



The 'A' TDRFP speed is being controlled by _____(1)_____.

Operator action is required to _____(2)_____ the 'A' TDRFP Min. flow valve to maintain system stability.

A.	(1) the Master Level Controller (2) open

B.	(1) the Master Level Controller (2) close

C.	(1) its hard controller (SLIM) (2) open

D.	(1) its hard controller (SLIM) (2) close

Answer:	A

Answer Justification / Plausibility Statements

A is correct.

The graphic shows the 'A' TDRFP operating in the Auto-Setpoint control mode (FPA Auto Setpolnt). Per CPS 3103.01 Feedwater (FW) step 4.19, operation of the TDRFP on the Master Level Controller is synonymous with the Auto Setpoint Mode.

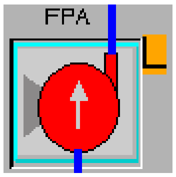
Per CPS 3103.01 Feedwater (FW), Precaution 4.19 and Limitation 6.15:

- TDRFP Auto Setpoint Mode response may NOT be stable below 2900 rpm. When shifting or feeding ensure >3000 rpm.
- May need to raise TDRFP Min. flow valve position to attain >3000 rpm.

With speed below 2900 rpm, there is an operational concern that the 'A' TDRFP response may not be stable Action must be taken to raise the 'A' TDRFP speed to > 3000 rpm. This can be achieved by opening the 'A' TDRFP Min. flow valve.

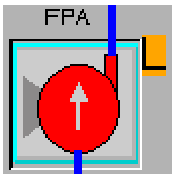
Incorrect Responses:

B is incorrect but plausible. Part 1 is correct. Part 2 is plausible because most systems are operated with flow above the minimum flow setpoint (keeping the min flow valve closed). Specifically, CPS 3309.01 High Pressure Core Spray (HPCS) precaution 4.2 (Pump Min Flow/Deadheading) states that low flow rates can cause hydraulic instability with the potential for increased pump wear. This limitation does not apply to the TDRFPs which can operate on minimum flow as long as turbine speed is maintained above 3000 RPM.
C is incorrect but plausible. This response would be correct if the 'A' TDRFP was operating in Local control mode on its hard controller (SLIM). The graphic below illustrates how the Ovation Digital Reactor Water Level Control system would depict the 'A' TDRFP operating in Local control mode. Part 2 is correct.



D is incorrect but plausible.

- Part 1 - This response would be correct if the 'A' TDRFP was operating in Local control mode on its hard controller (SLIM). The graphic below illustrates how the Ovation Digital Reactor Water Level Control system would depict the 'A' TDRFP operating in Local control mode.



- Part 2 is plausible because most systems are operated with flow above the minimum flow setpoint (keeping the min flow valve closed). Specifically, CPS 3309.01 High Pressure Core Spray (HPCS) precaution 4.2 (Pump Min Flow/Deadheading) states that low flow rates can cause hydraulic instability with the potential for increased pump wear. This limitation does not apply to the TDRFPs which can operate on minimum flow as long as turbine speed is maintained above 3000 RPM.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
259002.K5.07	K5.07	2.7	2.7	2		2

System Name
Reactor Water Level Control System

Category Statement
Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM : (CFR: 41.5 / 45.3)

K/A Statement
Turbine speed control mechanisms: TDRFP

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	259002A.16 EVALUATE the following FEEDWATER CONTROL indications/responses and DETERMINE if the indication/ response is expected and normal. .1 RPV level .2 TDRFP speed .3 MDRFP valve position .4 Ovation controller lineup

	Q44 259002 K5.07

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the candidate must demonstrate knowledge of the operational implications of the TDRFP speed control mechanisms as they apply to the Reactor Water Level Control System in order to select the correct response.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and application level. The examinee must analyze the graphic presented in the stem to determine the status of the affected equipment and apply system knowledge to determine behavior of the Ovation Digital Feedwater Controls based on those plant conditions (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 3103.01 Rev. 31f		
Training Objective	259002.16 EVALUATE the following FEEDWATER CONTROL indications/responses and DETERMINE if the indication/ response is expected and normal.		
Previous NRC Exam Use	None		

45	ID: 2105287	Points: 1.00

The plant is operating at rated thermal power.

Fuel Building Exhaust monitor 1RIX-PR006A is out of service and in a trip status.

THEN, the AR/PR high alarm for Fuel Building Exhaust monitor 1RIX-PR006D is received.

The expected total flow from the Standby Gas Treatment (VG) System is _____ scfm.

A.	0
B.	2000
C.	4000
D.	8000

EXAMINATION ANSWER KEY

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Answer:	D

Answer Justification / Plausibility Statements

D is correct.

Per CPS 5140.63 AR/PR Annunciator - Fuel Building Exhaust 1RIX-PR006A, B, C, D Auto Actions, a trip of 1RIX-PR006A or B coincident with a trip of 1RIX-PR006C or D will cause automatic isolation of Group 19 Fuel Building Ventilation (VF) and automatic start of Standby Gas Treatment (VG) A and B.

Per CPS 9067.01 Standby Gas Treatment System Train Flow/Heater Operability, step 2.1.3, VG has been designed with a flow control valve which operates on maintaining 4,000 CFM (+ / -10%) flow.

With both trains of VG running, expected total VG flow will be 4000 * 2 = 8000 CFM.

Incorrect Responses:

A is incorrect but plausible. This response would be correct if 1RIX-PR006B tripped. The logic would not be satisfied with PR006A and PR006B in a tripped condition.

B is incorrect but plausible. This response would be correct if the logic initiated only 1 VG train like other divisional trip systems such as ECCS, and the nominal flow for a single train was 2,000 CFM instead of 4,000 CFM.

C is incorrect but plausible. This response would be correct if the logic initiated only 1 VG train like other divisional trip systems such as ECCS.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
261000.K6.04	K6.04	2.9	3.1	2		9

System Name
Standby Gas Treatment System

Category Statement
Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : (CFR: 41.7 / 45.7)

K/A Statement
Process radiation monitoring

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q45 261000 K6.04

CPS.317442	DB497907.01.03 Describe the hazards created by a dropped fuel bundle.

Other NRC Data

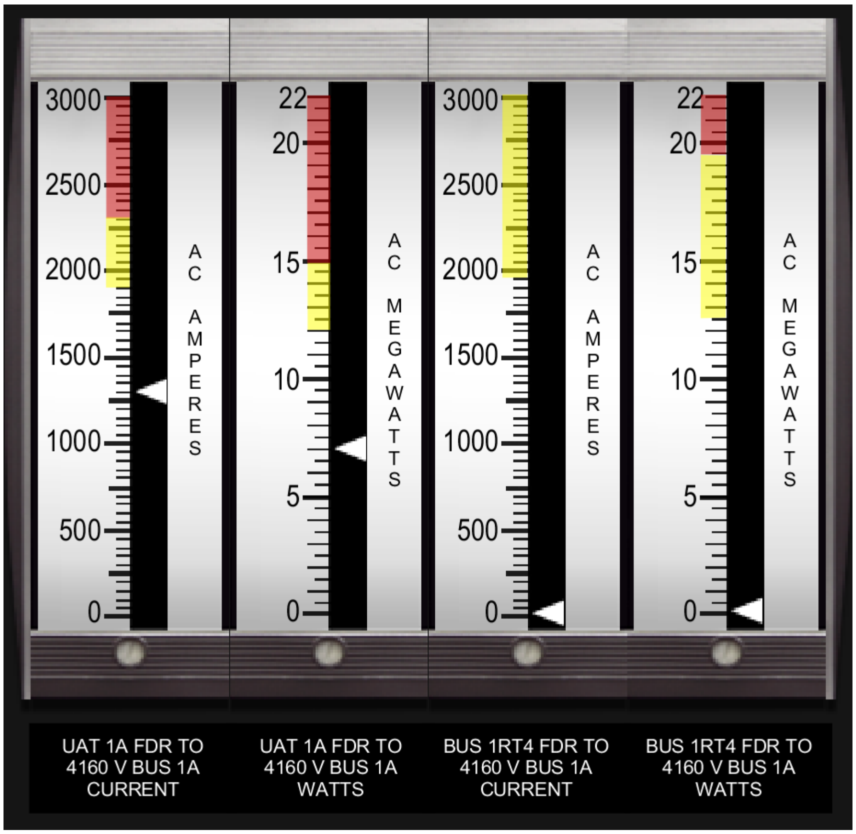
References Provided	None		
K/A Justification	This question meets the KA because the examinee must demonstrate knowledge of the effect that a loss or malfunction of a PRM will have on the VG system to determine the correct response.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall facts to select the correct response (1-F).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5140.63 Rev. 1c • CPS 9067.01 Rev. 31e		
Training Objective	261000.07 Given the VG SBGT STANDBY GAS TREATMENT system, DESCRIBE the systems supporting and the nature of the support.		
Previous NRC Exam Use	None		

46	ID: 2107249	Points: 1.00

The CRS has directed you to transfer 4160V Bus 1A to its reserve source IAW CPS 3501.01 High Voltage Power System, section 8.1.8 Transferring a 6900V or 4160V Bus TO or FROM its Reserve {Main} Source.

When the 4160V Bus 1A Res Bkr Sync keylock switch is placed in the ON position, the synchroscope will be _____(1)_____.

The following indications are observed after you take 4160V Bus 1A Res Bkr 1AP06EM control switch to the CLOSE position.



Required action is to _____(2)_____.

A.	(1) rotating (2) immediately release the 1AP06EM control switch

B.	(1) steady at ~ the 12 o'clock position (2) immediately release the 1AP06EM control switch

C.	(1) rotating (2) place the 4160V Bus 1A Res Bkr Sync keylock switch to OFF

D.	(1) steady at ~ the 12 o'clock position (2) place the 4160V Bus 1A Res Bkr Sync keylock switch to OFF

Answer:	D

Answer Justification / Plausibility Statements
D is correct: Part 1 CPS 3501.01 High Voltage Auxiliary Power System, step 8.1.8.1 directs the operator to verify the synchroscope is steady at ~ the 12 o'clock position. Part 2 CPS 3501.01 step 8.1.8.5 directs the operator to verify a load shift on the bus load meters. If the load shift is not indicated on the bus, then place the sync switch to OFF prior to releasing the switch to the AUTO position. With the Bus 1RT4 FDR to 4160 V Bus 1A Current and Watts meter indicating 0 amps and watts on the graphic in the stem, a load shift is not indicated and the sync switch should be returned to off prior to releasing the reserve breaker control switch. Incorrect Responses: A is incorrect but plausible. Part 1 - The synchroscope should be rotating when paralleling two different AC sources (such as a diesel generator with off site power). Part 2 - A caution in CPS 3501.01 before step 8.1.8.5 states that when transferring a bus, do not hold the Res Bkr control switch in CLOSE longer than 5 seconds to preclude an undesirable trip due to circulating currents between the transformers. B is incorrect but plausible. Part 1 is correct. Part 2 is plausible because a caution in CPS 3501.01 before step 8.1.8.5 states that when transferring a bus, do not hold the Res Bkr control switch in CLOSE longer than 5 seconds to preclude an undesirable trip due to circulating currents between the transformers. C is incorrect but plausible. Part 2 is correct. Part 1 is plausible because the synchroscope should be rotating when paralleling two different AC sources (such as a diesel generator with off site power).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
262001.A1.01	A1.01	3.1	3.4	2		6

System Name
A.C. Electrical Distribution

Category Statement
Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: (CFR: 41.5 / 45.5)

K/A Statement
Effect on instrumentation and controls of switching power supplies

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q46 262001 A1.01

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to predict the impact on synchroscope indications while operating AC Electrical Distribution controls to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to analyze conditions in the stem and then predict expected indications and determine required actions based on that analysis to answer the question (3-PEO/SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 3501.01 Rev. 29		
Training Objective	262001.11 EVALUATE given key Auxiliary Power System parameters, if needed DETERMINE a course of action to correct or mitigate the following abnormal condition(s): .8 Transferring a 6.9 or 4.16 kV Bus To/From its Reserve {Main} Source		
Previous NRC Exam Use	None		

47	ID: 2106727	Points: 1.00

The plant was operating at rated thermal power.

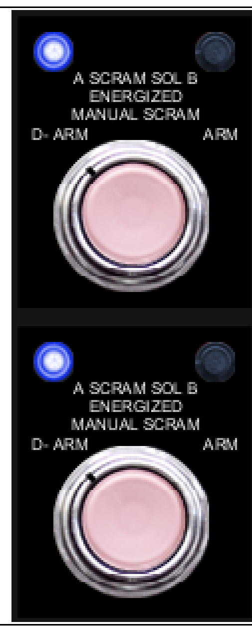
THEN, annunciator 5006-4L RPS SOLENOID INVERTER B TROUBLE was received.

The Equipment Operator reports that the OVER/UNDER VOLTAGE LED is LIT on the 'B' RPS Solenoid Inverter.

1) The status of Scram Solenoid Energized lights on Panel 1H13-P680-5004 AND 1H13-P680-5005 is depicted in...

Case 1

Case 2



2) In accordance with the annunciator response procedure for 5006-4L, the required operator action(s) is/are to _____(2)_____.

A.	(1) Case 1. (2) place transfer switch on 1C71-S004B to the BYPASS position ONLY

B.	(1) Case 1. (2) place transfer switch on 1C71-S004B to the BYPASS position AND shut RPS Solenoid Inverter B CB-3 breaker

C.	(1) Case 2. (2) place transfer switch on 1C71-S004B to the BYPASS position ONLY

D.	(1) Case 2. (2) place transfer switch on 1C71-S004B to the BYPASS position AND shut RPS Solenoid Inverter B CB-3 breaker

Answer:	D

Answer Justification / Plausibility Statements
D is correct. Part 1 Based on an inverter Over Voltage condition and per CPS 5006.04 Alarm Panel 5006 Annunciators - Row 4, Annunciator 5006-4L RPS SOLENOID INVERTER B TROUBLE: • The 'B' RPS Solenoid Inverter will trip on an Over voltage condition (as indicated by a lit OVER/UNDER VOLTAGE LED on the 'B' RPS Solenoid Inverter). • When the inverter trips, all B RPS scram solenoids will deenergize. A half scram will be indicated in the MCR (the 'B' scram solenoid energized lights extinguished at 1H13-P680). This is represented in the Case 2 graphic. Part 2 Per CPS 5006.04 (4L) and CPS 3509.01 Figure 3, required operator actions are to place the transfer switch on 1C71-S004B to the bypass position and shut CB-3 to reenergize the 'B' RPS solenoids. Incorrect Responses: A is incorrect but plausible: • Part 1 - This response would be correct if the trip of the 'B' RPS inverter caused one division of the RPS solenoids to deenergize. This would happen if RPS behaved like NSPS where loss of a divisional NSPS bus will result in deenergizing one of the four NSPS divisions and supplied loads. • Part 2 is plausible because restoration of NSPS loads requires reenergizing the associated NSPS bus from the alternate power source only. B is incorrect but plausible: • Part 1 - This response would be correct if the trip of the 'B' RPS inverter caused one division of the RPS solenoids to deenergize. This would happen if RPS behaved like NSPS where loss of a divisional NSPS bus will result in deenergizing one of the four NSPS divisions and supplied loads. • Part 2 is correct. C is incorrect but plausible: • Part 1 is correct. • Part 2 is plausible because restoration of NSPS loads requires reenergizing the associated NSPS bus from the alternate power source only.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
262002.A2.02	A2.02	2.5	2.7	2		6

System Name
Uninterruptable Power Supply (A.C./D.C.)

Category Statement
Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

K/A Statement
Over voltage

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

700004.15	700004.15 Given NSPS System initial conditions, PREDICT how the system and/or plant parameters will respond to the manipulation of the following controls. .1 NSPS Divisional Power manual transfer .2 Sync Loss light illumination during NSPS Divisional Power manual transfer .3 NSPS Divisional Power inverter control panel indications when Transfer Switch is in BYPASS position .4 NSPS Solenoid Bus Power manual transfer to normal or alternate power supply

	RO Q46 262002 A4.01

	Q47 262002 A2.02

Other NRC Data

References Provided	None		
K/A Justification	This question meets the A2(a) portion of the KA because the examinee has to demonstrate the ability to predict the impact of an overvoltage condition on a UPS (NSPS RPS Solenoid Inverter). As permitted by ES-401 D.2.a, the (b) portion of the K/A (the low cog portion) is not tested.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to predict how a system will respond to a NSPS RPS Solenoid Inverter overvoltage condition, and then determine the indications representing that prediction from a choice of 4 different graphics (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 5006.04 (4L) Rev. 30f		
Training Objective	700004.09 Discuss the effect: a. A total loss or malfunction of the NSPS System has on the plant. b. A total loss or malfunction of various plant systems has on the NSPS System. .4 Loss of NSPS Solenoid Power		
Previous NRC Exam Use	None		

48	ID: 2107177	Points: 1.00

Deenergizing DC MCC 1E will result in the immediate loss of...

A.	inverter input power for NSPS Solenoid Bus A.
B.	motor power to the Emergency Seal Oil Pump (ESOP).
C.	motor power to the Emergency Bearing Oil Pump (EBOP).
D.	control power to the Motor Driven Reactor Feed Pump (MDRFP).

Answer:	C

Answer Justification / Plausibility Statements
C is correct: Per CPS 3514.01C044 125VDC MCC 1E (1DC16E) BOP OUTAGE, step 2.1.2 states that the Emergency Bearing Oil Pump (EBOP) will lose power. Incorrect Responses: A is incorrect but plausible. Per CPS 3509.01 Figure 1, the Division 1 and 2 NSPS Inverters have a single DC supply to the inverter. The NSPS Solenoid Buses (Figure 3) have an inverter DC and AC power supply, so de-energizing DC MCC 1E will only result in a loss of the DC supply to the 'A' NSPS Solenoid Inverter. B is incorrect but plausible. Per CPS 3514.01C045 125VDC MCC 1F (1DC17E) BOP OUTAGE, step 2.1.1.B states that de-energizing DC MCC 1F (not DC MCC 1E) will result in a loss of power to the ESOP. D is incorrect but plausible. Per CPS 3103.01E001, 1FW01PC Rx Feed Pump 1C is powered from 6900V Bus 1B (1AP05E). Control power to 1AP05E is supplied from DC MCC 1F, not 1E.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
263000.K2.01	K2.01	3.1	3.4	2		6

System Name
D.C. Electrical Distribution

Category Statement
Knowledge of electrical power supplies to the following: (CFR: 41.7)

K/A Statement
Major D.C. loads

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

263000.08	263000.08 Given the BATTERY & DC DISTRIBUTION system, DESCRIBE the systems supported and the nature of the support. .1 Nuclear System Protection System (NSPS) .2 Reactor Core Isolation Cooling System (RCIC) .3 Auxiliary Power System .4 Remote Shutdown System .5 Turbine Lube Oil System .6 Seal Oil System .7 Diesel Generator System .8 Main Steam System .9 Switchyard System .10 Main Generator

263000.08	263000.08 Given the BATTERY & DC DISTRIBUTION system, DESCRIBE the systems supported and the nature of the support. .1 Nuclear System Protection System (NSPS) .2 Reactor Core Isolation Cooling System (RCIC) .3 Auxiliary Power System .4 Remote Shutdown System .5 Turbine Lube Oil System .6 Seal Oil System .7 Diesel Generator System .8 Main Steam System .9 Switchyard System .10 Main Generator

	Q48 263000 K2.01

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of electrical power supplies to major DC loads to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall facts about system power supplies to answer the question (1-F).		
NRC Exams Only			
Question Type	Bank (CL-ILT-4766)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3509.01 Rev. 22b • CPS 3514.01C044 Rev. 3b • CPS 3103.01E001 Rev. 13a • CPS 3514.01C045 Rev. 5d		
Training Objective	263000.08 Given the BATTERY & DC DISTRIBUTION system, DESCRIBE the systems supported and the nature of the support. .5 Turbine Lube Oil System		
Previous NRC Exam Use	None		

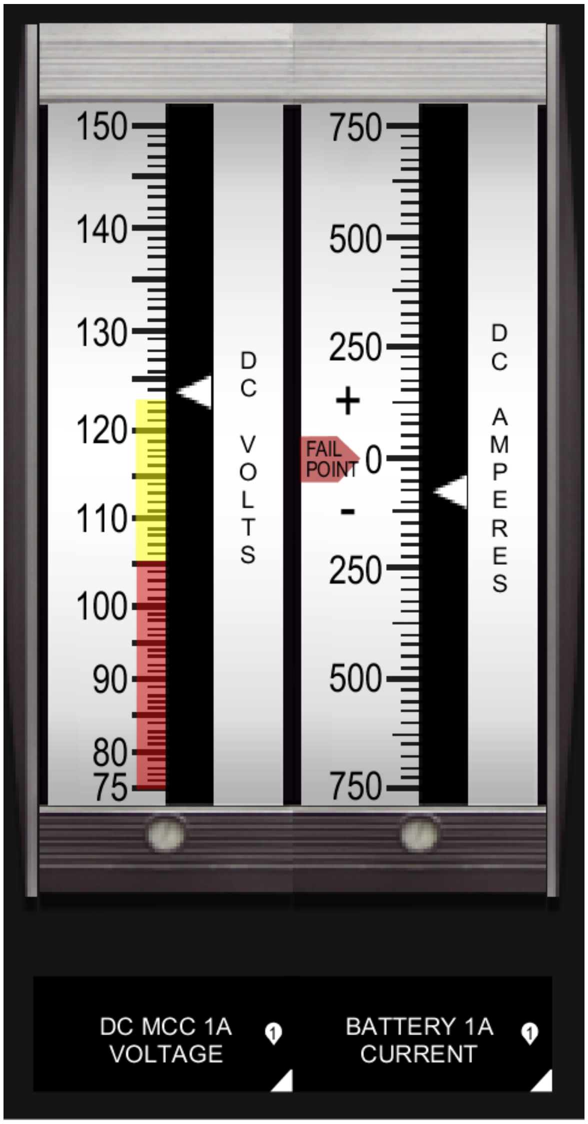
49	ID: 2106810	Points: 1.00

The plant was operating at rated thermal power.

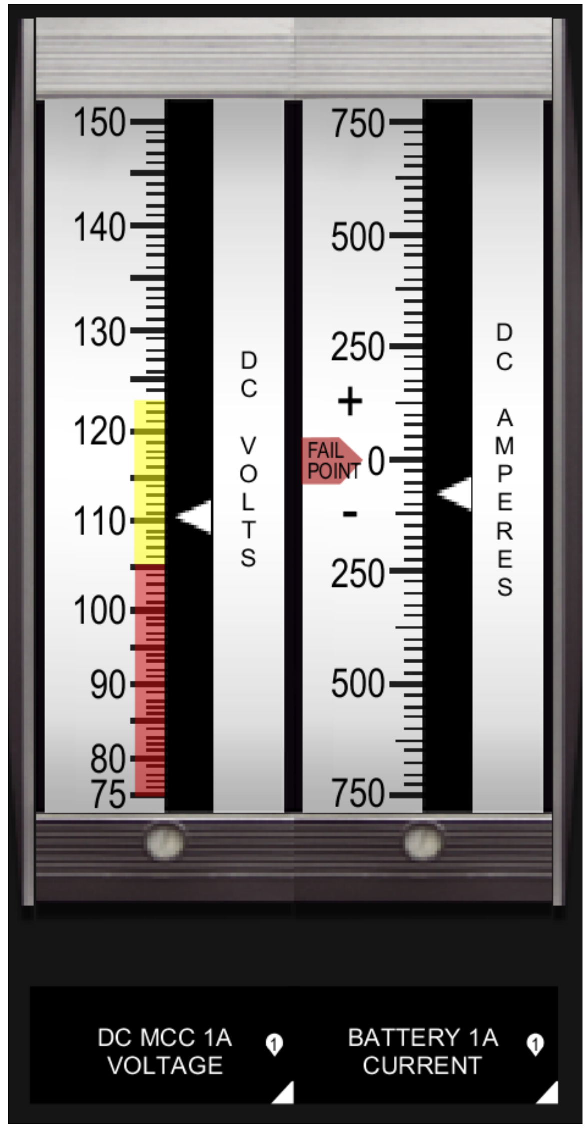
THEN, annunciator 5060-4E Trouble Batt Charger 1A was received due to actuation of the High DC Voltage Relay.

15 minutes later, the expected indications for DC MCC 1A Voltage and Battery 1A Current are shown in graphic _____(1)_____, and the status of annunciator 5060-2E Undervoltage 125V DC MCC 1A is shown in graphic _____(2)_____.

Graphic W



Graphic X



Graphic Y

Graphic Z

A.	(1) W (2) Y
B.	(1) W (2) Z
C.	(1) X (2) Y
D.	(1) X (2) Z

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Part 1 Per CPS 5060.04 (4E) Trouble Batt Charger 1A, high DC voltage (138V DC) shunt trips the AC input breaker on the battery charger, resulting in DC MCC 1A distribution panel being energized solely from it's respective battery. Immediately following the loss of the 1A Battery charger, DC MCC voltage will lower to ~ 125 VDC as Battery 1A begins to discharge - current will show negative amperes (current out of the battery). 15 minutes following the trip, DC MCC voltage does not decrease more than about 1 volt (~ 124-125 VDC). Part 2 Per CPS 5060-2E Undervoltage 125V DC MCC 1A, the alarm setpoint is 114V DC. With expected voltage 15 minutes after the trip of the battery charge at 124-125 VDC, 5060-2E should be OFF. Incorrect Responses: B is incorrect but plausible with DC MCC 1A Voltage lower than normal voltage (~132 VDC). Incorrect because the voltage in Graphic W is above the alarm setpoint. C is incorrect but plausible. DC MCC 1A Voltage will lower substantially when the Div 1 Battery Charger trips, but not to 110 VDC 15 minutes following the battery charger trip. The Div 1 Battery is sized to maintain voltage > 105 VDC for 4 hours if DC load shedding is completed within 1 hour of a loss of AC power to the battery charger. D is incorrect but plausible. • DC MCC 1A Voltage will lower substantially when the Div 1 Battery Charger trips, but not to 110 VDC 1 minute following the battery charger trip. The Div 1 Battery is sized to maintain voltage > 105 VDC for 4 hours if DC load shedding is completed within 1 hour of a loss of AC power to the battery charger. • Part 2 would be correct if DC voltage were at 110 VDC.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
263000.A3.01	A3.01	3.2	3.3	2		6

System Name
D.C. Electrical Distribution

Category Statement
Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: (CFR: 41.7 / 45.7)

K/A Statement
Meters, dials, recorders, alarms, and indicating lights

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q49 263000 A3.01

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must use graphics of DC Electrical Distribution system indications (meters and alarms) based on an automatic response to determine the correct response to the question.		
SRO-Only Justification	N/A		
Additional Information	Question is high cog, written at the analysis and comprehension level. The examinee must analyze the conditions provided in the stem and then select the graphics that correctly predicts the outcome of the event (3-PEO).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5060.04 (4E) Rev. 27 • CPS 5060.02 (2E) Rev. 26		
Training Objective	263000.16 EVALUATE the following BATTERY & DC DISTRIBUTION indications/responses and DETERMINE if the indication/ response is expected and normal.		
Previous NRC Exam Use	None		

50	ID: 2107606	Points: 1.00

The plant was operating at rated thermal power.

THEN, a transient resulted in the following conditions:

- Reactor scram
- Reactor water level lowered to -50 inches before recovering and rising at 2 inches/minute.

Which of the following cases describe the expected status of the Emergency Diesel Generators?

	Div 1 DG	Div 2 DG	Div 3 DG
Case 1	Standby	Standby	Running
Case 2	Running	Running	Running
Case 3	Standby	Standby	Standby
Case 4	Running	Running	Standby

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

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per CPS 5062.03 (3C) RUNNING DIESEL GEN 1C, a low reactor water level (-45.5") will result in the automatic start of the Div 3 DG. Incorrect Answers: B is incorrect but plausible. This answer would be correct if Div 1 and 2 DG start signals on low reactor water level were the same as the Div 3 DG. Per ITS 3.3.5.1, Table 3.3.5.1-1 Function 1A and 2A, RPV Level 1 provides initiation signals for the Div 1 and Div 2 DGs. C is incorrect but plausible. This answer would be correct if the RPV level initiation signal for Div 3 was the same as Div 1 and 2, and all 3 DGs automatically started at RPV Level 1. D is incorrect but plausible. This answer would be correct if Div 1 and Div 2 DGs started at RPV Level 2, and Div 3 DG started at RPV Level 1. The opposite is true.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
264000.K4.08	K4.08	3.8	3.7	2		6

System Name
Emergency Generators (Diesel/Jet)

Category Statement
Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K/A Statement
Automatic startup

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q50 264000 K4.08

Other NRC Data

References Provided	None		
K/A Justification	Question meets the KA because the examinee has to determine that the Div 3 DG automatically starts with the information provided in the stem to answer the question correctly.		
SRO-Only Justification	N/A		
Additional Information	Question is written at the memory level. The examinee has to recall the automatic start signals for the Div 3 DG to answer the question (1-I).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5062.03 (3C) Rev. 30c • CPS 5010.04 (4B) Rev. 28 • CPS 5010.01 (1A) Rev. 27 • ITS 3.3.5.1 (3.3-39 and 3.3-40) Amendment 216		
Training Objective	264000.05 Discuss the DIESEL GENERATOR/DIESEL FUEL OIL system automatic functions/interlocks including purpose, signals, set points, sensing points, when bypassed, how/when they are. .2 Diesel Generators		
Previous NRC Exam Use	None		

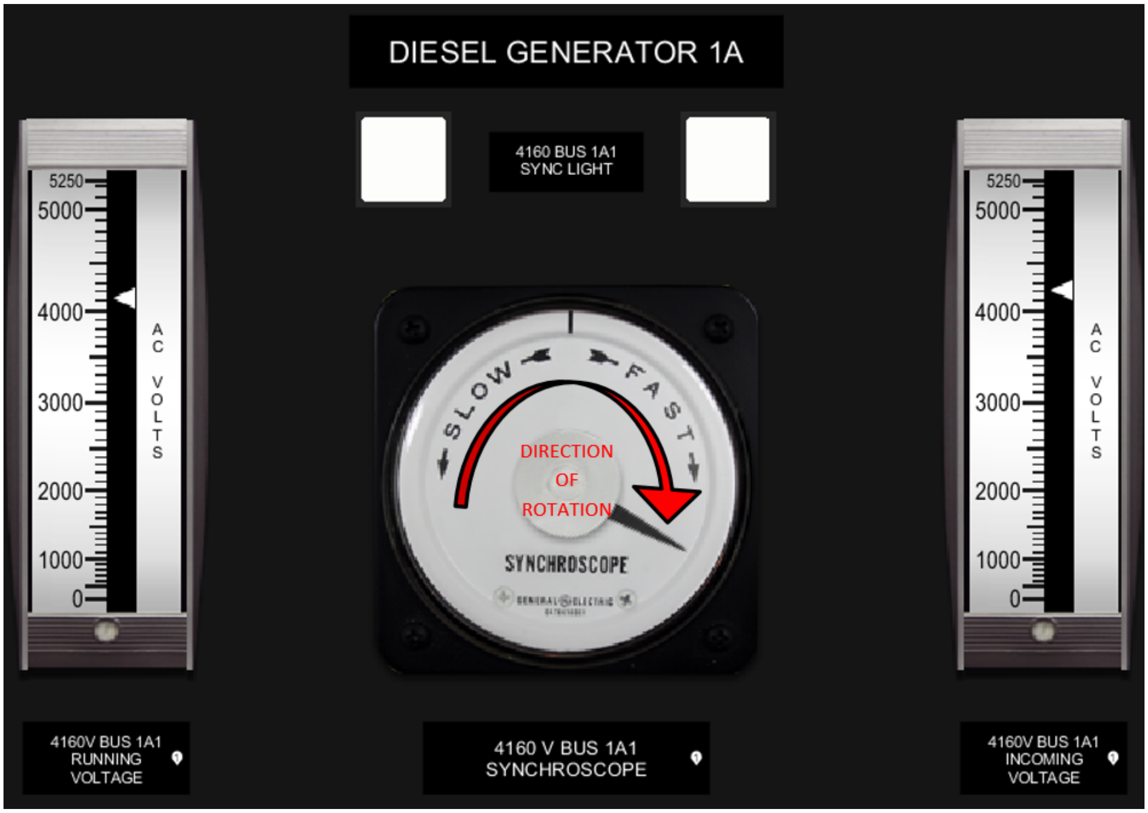
51	ID: 2107130	Points: 1.00

The plant was operating at rated thermal power when a loss of offsite power occurred.

Offsite power has been restored.

The CRS has directed you to transfer 4160 Volt bus 1A1 from DG1A to the Reserve Aux Transformer.

Indications are as follows:



Which of the following control switch manipulations must be performed to synchronize DG1A and the RAT?

DG1A Voltage Regulator switch in the ____(1)____ direction.

DG1A Governor switch in the ____(2)____ direction.

A.	(1) Raise (2) Raise
B.	(1) Raise (2) Lower
C.	(1) Lower (2) Raise
D.	(1) Lower (2) Lower

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per CPS 3506.01 Diesel Generator and Support Systems (DG), 8.4.3 Transferring 4160V Bus Power From DG To RAT 'B' (ERAT): Match Running Voltage to Incoming voltage: • Based on the graphic, INCOMING voltage (RAT 'B') is currently higher than RUNNING voltage (DG1A). • DG1A voltage must be raised to match RAT 'B' voltage. • The operator will bump the DG1A Voltage Regulator switch in the <u>Raise</u> direction until RUNNING voltage is matched with INCOMING voltage. Adjust DG1A speed such that DG frequency is slightly higher than RAT 'B' frequency as indicated by a COUNTER-CLOCKWISE rotation: • Based on the graphic, the synchroscope is currently turning in the FAST direction (CLOCKWISE). • DG1A frequency must be increased until it is slightly higher than RAT 'B' frequency. • The operator will bump the DG1A Governor switch in the <u>Raise</u> direction until the synchroscope is rotating slowly in the SLOW direction (COUNTER-CLOCKWISE). Incorrect Responses: B is incorrect but plausible. This response would be correct if COUNTER-CLOCKWISE rotation of the synchroscope was achieved when DG1A frequency is lower than RAT 'B' frequency and the operator lowered DG1A speed to meet the requirement for generator synchronization. However, per CPS 3506.01, the DG1A speed must be adjusted to make DG1A frequency slightly higher than RAT 'B' frequency. C is incorrect but plausible. This response would be correct if the DG1A voltage was indicated as the Incoming voltage and the operator lowered the Incoming voltage to match the Running voltage. However, per CPS 3506.01, the Running voltage (DG1A) is adjusted to match the Incoming voltage (RAT 'B'). D is incorrect but plausible. This response would be correct if: • the DG1A voltage was indicated as the Incoming voltage and the operator lowered the Incoming voltage to match the Running voltage, and • COUNTER-CLOCKWISE rotation of the synchroscope was achieved when DG1A frequency is lower than RAT 'B' frequency and the operator lowered DG1A speed to meet the requirement for generator synchronization.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
264000.A4.05	A4.05	3.6	3.7	2		6

System Name
Emergency Generators (Diesel/Jet)

Category Statement
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

K/A Statement
Transfer of emergency generator (with load) to grid

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q51 264000 A4.05

Other NRC Data

References Provided	None
K/A Justification	Question meets the KA because the examinee must demonstrate knowledge of the ability to manually transfer a DG (with load) to the grid (RAT 'B') from the MCR to select the correct response.
SRO-Only Justification	N/A
Additional Information	Question is written at the comprehension and application level. The examinee has to analyze the graphic presented in the stem, determine the alignments of two AC sources (DG1A & RAT 'B') and select what steps must be taken to parallel those two sources (3-SPK).

NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 3506.01 Rev. 39c		
Training Objective	264000.04 STATE the physical location and function of the following DIESEL GENERATOR/DIESEL FUEL OIL system components, controls, indicators, and/or sensors. .9 Governor Controls		
Previous NRC Exam Use	None		

52	ID: 2105239	Points: 1.00

The plant was operating at Rated Thermal Power.

THEN, the following annunciators were received:

- 5041-6B Auto Start Service Air Compressor
- 5042-4D Trouble VF System 1PL44J

The 'B' RO reports:

- Service Air Compressor '0' has auto started, AND
- 0SA01C and 1SA01C are operating at 80 amps, AND
- amps are slowly lowering for both Service Air Compressors
- Fuel Building Ventilation (VF) system has shutdown
- Service Air Header Pressure lowered and then rose to 105 psig and is rising slowly.

Which of the following describes the plant/system impact of this event?

Ring Header Isolation Valves __________ have auto closed.

A.	1IA022 AND 1IA021
B.	1IA045 AND 1IA021
C.	1SA037 AND 1SA001
D.	1SA059 AND 1SA038

Answer:	A

Answer Justification / Plausibility Statements
A is correct. Per CPS 5041.04 Alarm Panel 5041 Annunciators - Row 4, LOW PRESS CONTROL BLDG IA RING HDR (5041-4C): • A leak in the CB IA ring header will result in automatic closure of 1IA022 & 1IA021 Control Bldg IA Ring Header Isolation Valves, and • a loss of instrument air to the loads supplied from that ring header (including Fuel Building Ventilation (VF). With SA compressor amperage high but lowering and Service Air Header Pressure low but rising, the examinee should conclude that the leak has been isolated. Additionally, per 5041-4C a loss of instrument air to the loads supplied from the Control Building IA ring header include: • Following HVAC systems shutdown: CNMT/Drywell Purge (VR/VQ), **Fuel Bldg (VF)**, Aux Bldg (VA), Turbine Bldg (VT), Machine Shop (VJ), Laboratory (VL), Radwaste Bldg (VW), Makeup Air portion of Diesel Generator (VD). • Plant Chilled Water (WO) Chillers shutdown. The makeup valve to the WO Compression Tank fails open & lifts the WO relief valve. • CCW Storage Tank Automatic Makeup Water Valve will fail closed. Incorrect Responses: B is incorrect but plausible. This response would be correct for a low pressure condition in the Radwaste Building IA Ring Header, however the Fuel Building Ventilation system is unaffected by a loss of RW Bldg IA. C is incorrect but plausible. This response would be correct for a low pressure condition in the Radwaste Building Service Air (SA) Ring Header, however the Fuel Building Ventilation system is unaffected by a loss of RW Bldg SA. D is incorrect but plausible. This response would be correct for a low pressure condition in the Control Building SA Ring Header, however the Fuel Building Ventilation system is unaffected by a loss of Control Bldg SA.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.300000	B2.4.47	4.2	4.2	2	1	N/A

System Name
Instrument Air System (IAS)

Category Statement
Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (CFR: 41.10 / 43.5 / 45.12)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q52 300000 2.4.47

300000.08	300000.08 Given the Service and Instrument Air system, DESCRIBE the systems supported and the nature of the support.

Other NRC Data

References Provided	CPS 3214.01 Figure 1 Service and Instrument Air Diagram		
K/A Justification	This question meets the KA because the examinee must diagnose and recognize trends in the IA system utilizing control room reference material to answer the question correctly.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to interpret the indications provided in the stem of the question and using control room reference material determine the system impact to answer the question (3-SPK/SPR).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A12032)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3214.01 Rev. 27c • CPS 5041.04 (4B, 4C) Rev. 27 • CPS 5041.05 (5B, 5C) Rev. 29a		
Training Objective	300000.08 Given the Service and Instrument Air system, DESCRIBE the systems supported and the nature of the support.		
Previous NRC Exam Use	None		

53	ID: 2105163	Points: 1.00

1RIX-PR037 Component Cooling Water Process Radiation Monitor is physically connected to the Component Cooling Water (CCW) system at the _____(1)_____.

If radiation levels are rising on 1RIX-PR037, a possible source of inleakage is a tube leak in a _____(2)_____.

A.	(1) CCW Pump Suction Header (2) RWCU Non-Regenerative Heat Exchanger ONLY

B.	(1) CCW Header inside the containment (2) RWCU Non-Regenerative Heat Exchanger ONLY

C.	(1) CCW Pump Suction Header (2) FC Heat Exchanger or RWCU Non-Regenerative Heat Exchanger

D.	(1) CCW Header inside the containment (2) FC Heat Exchanger or RWCU Non-Regenerative Heat Exchanger

Answer:	C

Answer Justification / Plausibility Statements

C is correct

Per CPS 4979.05 Abnormal Release of Radioactive Liquids, step 4.6.3, primary sources of radioactivity may include:

- RT Pump seal coolers
- RR Pumps
- RT NRHXs
- Reactor Sample Station
- FC HXs

Per CPS 3203.01 Component Cooling Water (CC):

- None of these loads (with the exception of the FC HXs), are monitored directly by Process Radiation Monitors on the CC returns.
- A radioactive leak into the CC system would be indicated by Process Radiation Monitor 1RIX-PR037 which samples the common CC return header upstream of the CCW Pumps.
- Because the CCW system is a closed loop system, a tube leak on any of the listed loads will cause 1RIX-PR037 readings to rise. This includes the Fuel Pool Heat Exchangers which are redundantly monitored by 1RIX-PR004(5) Fuel Pool Heat Exchangers 1A(1B) Service Water Effluent PRMs.

Incorrect Responses:

A is incorrect but plausible. This response would be correct if the CCW system operated at a higher pressure than the FC system such that a FC HX tube leak would not be indicated on 1RIX-PR037.

B is incorrect but plausible. This response would be correct if:

- 1RIX-PR037 sampled from the CCW Return Header inside the containment closest to the systems cooled by CCW with the highest levels of radioactivity, and
- the CCW system operated at a higher pressure than the FC system such that a FC HX tube leak would not be indicated on 1RIX-PR037.

D is incorrect but plausible. This response would be correct if 1RIX-PR037 sampled from the CCW Return Header inside the containment closest to the systems cooled by CCW with the highest levels of radioactivity.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
400000.K1.03	K1.03	2.7	3.0	2		8

System Name
Component Cooling Water System (CCWS)

Category Statement
Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K/A Statement
Radiation monitoring systems

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.2	41.2
41.3	41.3
41.4	41.4
41.5	41.5
41.6	41.6
41.7	41.7
41.8	41.8
41.9	41.9

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q53 400000 K1.03

272000.03	272000.03 DESCRIBE the function, operation, interlocks, trips, and power supplies of the following AR/PR System components. Area Radiation Monitors (ARMs) Continuous Air Monitors (CAMs) Main Steam Line Radiation Monitors (MSLRMs) Containment Building Continuous Containment Purge (CCP) Duct Monitors Containment Building Fuel Transfer Pool Vent Plenum Monitors Containment Building Exhaust Duct Monitors Fuel Building (FB) Exhaust Vent Plenum Monitors Main Control Room Air Intake Monitors HVAC Exhaust Stack Monitors 0 SGTS Exhaust Stack Monitors 1 Pre-Treatment Offgas Monitor 2 Post Treatment Offgas Monitors 3 HVAC Accident Range (AXM) Monitor 4 SGTS Accident Range (AXM) Monitor 5 Fuel Pool Cooling & Cleanup (FC) Heat Exchanger (HX) 1A (1B) WS Effluent Monitors 6 Plant Service Water (WS) Effluent Monitor 7 Component Cooling Water (CC) HX Monitor 8 Shutdown Service Water (SX) Effluent Monitors 9 Liquid Radwaste Discharge Monitor

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must have knowledge of the physical connections and the cause-effect relationships between the CCW System and the Radiation Monitoring System to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall facts to select the correct response (1-F).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A12028)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3203.01 Rev. 35e • CPS 4979.05 Rev. 9b		
Training Objective	400001.07 Given the Component Cooling Water system, DESCRIBE the systems supporting and the nature of the support.		
Previous NRC Exam Use	None		

54	ID: 2105011	Points: 1.00

The plant was operating at rated thermal power.

THEN, annunciator 5006-1H ACCUMULATOR TROUBLE alarms.

The Reactor Operator will depress the ____(1)____ button to determine the affected Hydraulic Control Unit (HCU).

An Equipment Operator is dispatched to investigate and reports the affected HCU pressure is 1680 psig.

The affected HCU must be ____(2)____.

A.	(1) ACCUM FAULT (2) recharged

B.	(1) ACCUM FAULT (2) drained of excess water

C.	(1) ACKN ACCUM FAULT (2) recharged

D.	(1) ACKN ACCUM FAULT (2) drained of excess water

Answer:	B

Answer Justification / Plausibility Statements

B is correct.
 	
Per CPS 5006.01 Alarm Panel 5006 Annunciators - Row 1, ACCUMULATOR TROUBLE (5006-1H):
 	
• Depress ACCUM FAULT button - the red status light for the rod associated with the alarming Hydraulic Control Unit (HCU) will flash.
• After the alarming HCU has been identified - acknowledge ACCUM FAULT on OCM to clear annunciator so another fault will cause an alarm.
• Verify control rod scram accumulator pressure locally:
    • If pressure is low (< 1620 psig), recharge the HCU per CPS 3304.01 Control Rod Hydraulic And Control (RD).
    • If pressure is normal (> 1620 psig), then drain excess water per CPS 3304.01.
 	
Incorrect Responses:
 	
A is incorrect but plausible. This response would be correct if the affected HCU pressure was < the minimum accumulator setpoint value (< 1620 psig). However, since the affected HCU pressure was reported at ~ normal operating pressure, it can be concluded that the accumulator level switch initiated the ACCUMULATOR TROUBLE annunciator.
 	
C is incorrect but plausible. This response would be correct if depressing the ACKN ACCUM FAULT pushbutton on the OCM allowed the operator to determine the affected HCU. However, this action will clear the current annunciator so a following fault will cause the annunciator to reinitiate.
 	
D is incorrect but plausible. This response would be correct if:
• depressing the ACKN ACCUM FAULT pushbutton on the OCM allowed the operator to determine the affected HCU, and
• the affected HCU pressure was < the minimum accumulator setpoint value.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
201001.A3.07	A3.07	3.3	3.3	2		1

System Name
Control Rod Drive Hydraulic System

Category Statement
Ability to monitor automatic operations of the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: (CFR: 41.7 / 45.7)

K/A Statement
HCU accumulator pressure/level

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

201001.02	201001.02 DESCRIBE the major flowpaths for the following modes of the Control Rod Drive Hydraulic System operation. .1 Suction Flowpath .2 Discharge Flowpath .3 Control Rod Insertion .4 Control Rod Withdrawal .5 Reactor Scram

	Q54 201001 A3.07

201001.16	201001.16 EVALUATE the following Control Rod Drive Hydraulic indications/responses and DETERMINE if the indication/ response is expected and normal. .1 Reactor Scram

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must demonstrate the ability to monitor automatic operation of the RD system including HCU accumulator pressure/level to determine the correct response.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall procedure steps to select the correct response (1-P).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 5006.01 (1H) Rev. 32e		
Training Objective	201001.06 Given an Control Rod Drive Hydraulic System Annunciator, DESCRIBE: a. The condition causing the annunciator b. Any automatic actions c. Any operational implications		
Previous NRC Exam Use	None		

55	ID: 2107628	Points: 1.00

The plant is operating at rated thermal power.

CPS 9813.01 CONTROL ROD SCRAM TIME TESTING is in progress on control rod 16-53.

1. In which case would the SCRAM VALVES pushbutton on the 1H13-P680 Display Selection Matrix first illuminate?

	NORM-TEST-SRI Toggle Switch 1	NORM-TEST-SRI Toggle Switch 2
Case 1	Test	Normal
Case 2	Test	Test

2. An illuminated SCRAM VALVES pushbutton means that both scram pilot valve solenoids are...

A.	(1) Case 1 (2) energized.

B.	(1) Case 2 (2) energized.

C.	(1) Case 1 (2) deenergized.

D.	(1) Case 2 (2) deenergized.

Answer:	D

Answer Justification / Plausibility Statements

D is correct:

Per CPS 3304.02 Rod Control and Information System (RC&IS) section 8.1.8.5 Scram Valves - the pushbutton will be backlit to indicate that at least 1 rod has a scram valve whose position is different from the other scram valves. At rated thermal power, the button is expected to be dark indicating that the insert and exhaust scram valves on the HCU are shut.

Per CPS 3304.01 Control Rod Hydraulic and Control (RD), step 8.2.7.7, both NORM-TEST-SRI toggle switches have to be placed in test to open the associated HCU Scram Inlet and Outlet valves.

Per ITS B3.3.1.1 RPS Instrumentation, bases page 3.3-2 states that two scram pilot valves are located in the hydraulic control unit for each control rod drive. Each scram pilot valve is solenoid operated, with the solenoids normally energized.

Incorrect Responses:

A is incorrect but plausible. Part 1 is incorrect but plausible and would be correct if the SRI toggle switches were wired in series such that placing either switch in test would result in de-energization of the scram pilot valve solenoids. Part 2 is incorrect but plausible and would be correct if the scram pilot valve solenoids operated like the backup scram valve solenoids. The backup scram valve solenoids energize on a scram signal to depressurize the scram air header.

B is incorrect but plausible. This answer would be correct if the scram pilot valve solenoids operated like the backup scram valve solenoids. The backup scram valve solenoids energize on a scram signal to depressurize the scram air header.

C is incorrect but plausible. Part 1 is incorrect but plausible and would be correct if the SRI toggle switches were wired in series such that placing either switch in test would result in de-energization of the scram pilot valve solenoids.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
201005.A4.02	A4.02	3.7	3.7	2		1

System Name
Rod Control and Information System (RCIS)

Category Statement
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

K/A Statement
Rod display module (lights and push buttons): BWR-6

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

201002.02	201002.02 DESCRIBE the major flowpaths for the following modes of the RC&IS System operation. .1 Rod Insertion .2 Rod Withdrawal .3 Signal Flowpath

	Q55 201005 A4.02

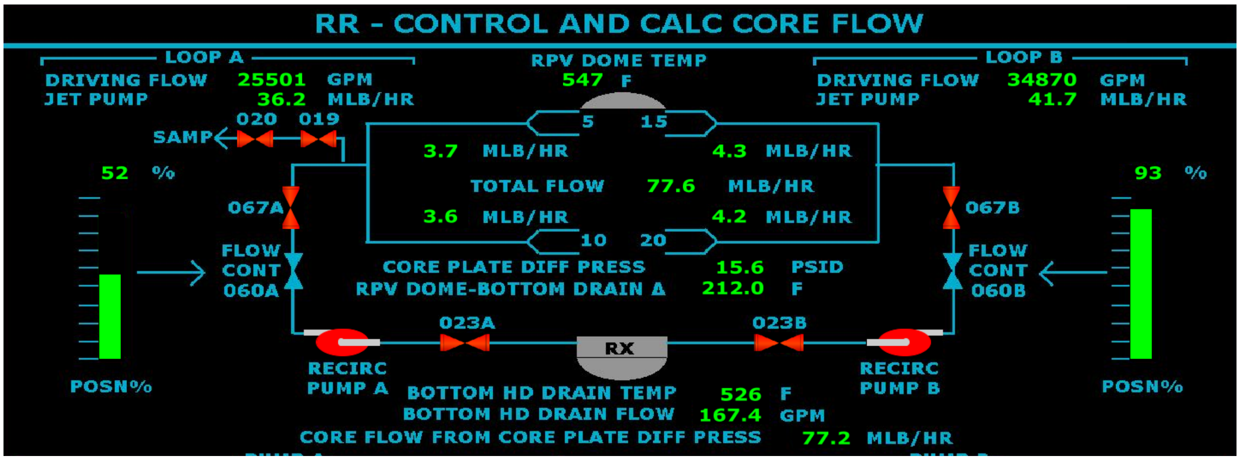
Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must demonstrate the ability to monitor RC&IS Rod Display Module lights and pushbuttons to determine the correct response.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall the function of system lights and components to select the correct response (1-F).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3304.01 Rev. 38 • CPS 3304.02 Rev. 22e • ITS B 3.3.1.1 (B 3.3-2) Rev. 10-6		
Training Objective	201001.03 DESCRIBE the function, operation, interlocks, trips, physical locations and power supplies of the following Control Rod Drive Hydraulic System components. .23 HCU SCRAM Pilot Solenoids/Pilot Valve		
Previous NRC Exam Use	None		

56	ID: 2106900	Points: 1.00

The plant was operating at rated thermal power.

THEN, a transient occurred resulting in the indications shown below:



Reactor power is stable at 92.6%.

Loop flow mismatch is _____(1)_____ the limit of ITS 3.4.1 Recirculation Loops Operating.

Current core flow indications _____(2)_____ entry into LCOs 3.2.1 Average Planar Linear Heat Generation Rate (APLHGR), 3.2.2 Minimum Critical power ratio (MCPR), and 3.2.3 Linear Heat Generation Rate (LHGR).

A.	(1) less than (2) require

B.	(1) less than (2) do NOT require

C.	(1) greater than (2) require

D.	(1) greater than (2) do NOT require

Answer:	D

Answer Justification / Plausibility Statements
D is correct.

Per ITS SR 3.4.1.1 recirculation loop jet pump flow mismatch with both recirculation loops in operation must be verified less than or equal to:

- 10% of rated core flow when operating at less than or equal to 70% of rated core flow (.7 * 84.5 = 59.15 mlbm/hr, or
- less than or equal to 5% of rated core flow when operating at > or equal to 70% of rated core flow (59.15 mlbm/hr)

With total core flow at 77.6 mlbh, the loop flow mismatch is limited to 5% of 84.5 mlbh (4.225 mlbm/hr).

The actual loop flow mismatch is 41.7 - 36.2 = 5.5 mlbm/hr which is <u>above</u> the loop flow mismatch limit of ITS 3.4.1.

Per CPS 3005.01 Unit Power Changes and by applying the appropriate plant parameters to Figure 1: Stability Control & Power/Flow Operating Map:

- The plant is operating <u>below</u> the MELLLA limit
- Entry into LCOs 3.2.1 Average Planar Linear Heat Generation Rate (APLHGR), 3.2.2 Minimum Critical power ratio (MCPR), and 3.2.3 Linear Heat Generation Rate (LHGR) is <u>not</u> required.

Incorrect Responses:

A is incorrect but plausible. This answer would be correct if:

- the loop flow mismatch was limited to 10% of 84.5 mlbh (8.45 mlbm/hr), and
- plant parameters indicate that the plant is operating above the MELLLA limit.

B is incorrect but plausible. This answer would be correct if the loop flow mismatch was limited to 10% of 84.5 mlbh (8.45 mlbm/hr).

C is incorrect but plausible. This answer would be correct if plant parameters indicate that the plant is operating above the MELLLA limit.

<u>K/A Data</u>

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.202001	B2.2.42	3.9	4.6	2	2	N/A

System Name
Recirculation System

Category Statement
Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
43.2	43.2
43.3	43.3

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q56 202001 2.2.42

Other NRC Data

References Provided	CPS 3005.01. Figure 1		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to recognize system parameters that are entry-level conditions for Technical Specifications to answer the question.		
SRO-Only Justification	N/A. Reactor operators are responsible for verifying SR 3.4.1.1 loop flow mismatch within limits by performing CPS 9000.01D001 Control Room Surveillance Log – Mode 1, 2, and 3 Data Sheet, section 8.24 Recirculation Loop Flow. This is required knowledge for ROs at CPS.		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the indication provided in the stem and then determine which annunciators should be in alarm to answer the question (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 3005.01 Rev. 43f		
Training Objective	LP87437.01.03 Given a Core Operating Map, core flow, reactor power and a specified change in power or flow; predict the final operating point on the map.		
Previous NRC Exam Use	None		

57	ID: 2105002	Points: 1.00

At rated thermal power, the Reactor Water Cleanup System normally takes a suction from...

A.	RR Loop 'A' AND 'B' ONLY
B.	RR Loop 'A' AND RPV Bottom Head Drain ONLY
C.	RR Loop 'B' AND RPV Bottom Head Drain ONLY
D.	RR Loop 'A' AND 'B', and the RPV Bottom Head Drain

Answer:	D

Answer Justification / Plausibility Statements

D is correct:
 	
Per CPS 3303.01 REACTOR WATER CLEANUP (RT), step 6.11.1, RT system flow should be adjusted to maintain the following limits:
 	
• In Modes 1, 2, and 3, to prevent thermal stratification of the Bottom Head region and RR suction lines:
  • Maintain **Bottom Head Drain** flow ≥ 63 gpm
  • Maintain RWCU suction from each **non-isolated RR loop** ≥ 30 gpm. RR loop flows to RWCU can be determined using system flow and bottom head drain flow.
 	
Incorrect Responses:
 	
A is incorrect (RT takes suction from RR 'A', 'B', and the RPV Bottom Head Drain) but plausible because 1G33-F101 RT Bottom Head Drain Suction motor operated valve is normally closed in Mode 1. Flow through the bottom head drain is maintained by 1G33-F102, RWCU Recirc Suct Throt, which is manually operated and whose position is not indicated in the MCR.
 	
B and C are incorrect but plausible and would be correct if the RT system was configured like the RHR Shutdown Cooling subsystems which take suction from a single RR loop at CPS.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
204000.K1.02	K1.02	2.9	3.0	2		2

System Name
Reactor Water Cleanup System

Category Statement
Knowledge of the physical connections and/or causeeffect relationships between REACTOR WATER CLEANUP SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K/A Statement
Recirculation system: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.2	41.2
41.3	41.3
41.4	41.4
41.5	41.5
41.6	41.6
41.7	41.7
41.8	41.8
41.9	41.9

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q57 204000 K1.02

DB400601.01.05	DB400601.01.05 Discuss when Reactor Water Cleanup (RT) may be used for preventing thermal stratification of the reactor coolant or for decay heat removal.

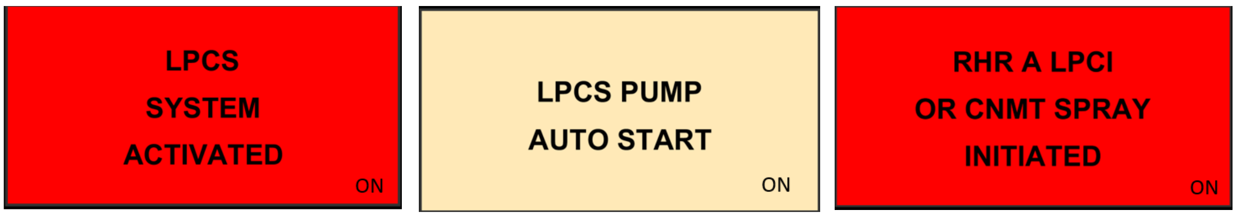
Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must have knowledge of the physical connections between Reactor Water Cleanup (RWCU) system and the Reactor Recirculation (RR) system to answer the question.		
SRO-Only Justification	N/A		
Additional Information	Question is written at the memory level (low cog). The examinee has to recall the physical connections between Reactor Water Cleanup (RWCU) system and the Reactor Recirculation (RR) system (1-P).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 3303.01 Rev. 37b		
Training Objective	204000.02 DESCRIBE the major flowpaths for the following modes of the RWCU System operation. .1 Normal Operations		
Previous NRC Exam Use	N/A		

58	ID: 2107260	Points: 1.00

The plant was operating at rated thermal power with both loops of RHR running in **FULL** Suppression Pool Cooling Mode.

THEN, the following alarms and indications were observed:



If directed to realign both RHR loops into **FULL** Suppression Pool Cooling Mode, 1E12-F024A & F024B RHR A(B) Test Valve To Suppr Pool can be opened _____(1)_____, and 1E12-F048A & F048B RHR A(B) Hx Bypass Valve can be reclosed _____(2)_____.

A.	(1) immediately (2) immediately
B.	(1) immediately (2) in ten minutes
C.	(1) in ten minutes (2) immediately
D.	(1) in ten minutes (2) in ten minutes

Answer:	B

Answer Justification / Plausibility Statements

B is correct:

Per CPS 5064.02 (2G) RHR A LPCI OR CNMT SPRAY INITIATED and 5065.02 (2D) RHR B/C LPCI OR CNMT SPRAY INITIATED, a LPCI initiation signal will result in RHR Loops 'A' and 'B' automatically shifting from Suppression Pool Cooling to Low Pressure Coolant Injection Mode (a loss of SP Cooling).

Per CPS 3312.01 Residual Heat Removal (RHR), section 8.1.2 LPCI Automatic Initiation, if RHR was not in standby when LPCI initiated, verify following valves automatically repositioned as follows:

- **1E12-F048A(B), RHR A(B) Hx Bypass Valve - OPEN**
- **1E12-F024A(B), RHR A(B) Test Valve To Suppr Pool - Shut**
- 1E12-F021, RHR C Test Valve To Suppr Pool - Shut
- 1E12-F028A(B), RHR A(B) To CNMT Spray A(B) Shutoff Vlv - Shut
- 1E12-F027A(B), RHR A(B) To CNMT Outbd Isol Valve - OPEN

Per CPS 3312.01 section 8.1.11 Suppression Pool Cooling - Shifting From LPCI, step 8.1.11.3 directs opening 1E12-F024A(B). A note above the step for opening 1E12-F048A(B) states that 1E12-F048A(B) will open for 10 minutes after receipt of a LPCI signal, and then can be repositioned.

Incorrect Responses:

A is incorrect but plausible. Part 1 is correct (1E12-F024A(B) can be opened immediately). Part 2 would be correct if 1E12-F048A(B) operated similarly to 1E12-F024A(B).

C is incorrect but plausible and would be correct if 1E12-F024A(B) operated similarly to 1E12-F048A(B) and if 1E12-F048A(B) operated similarly to 1E12-F024A(B).

D is incorrect but plausible. Part 2 is correct. Part 1 would be correct if 1E12-F024A(B) operated similarly to 1E12-F048A(B).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
219000.K3.01	K3.01	3.9	4.1	2		5

System Name
RHR/LPCI: Torus/Suppression Pool Cooling Mode

Category Statement
Knowledge of the effect that a loss or malfunction of the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE will have on following: (CFR: 41.7 / 45.4)

K/A Statement
Suppression pool temperature control

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

203000.15	203000.15 Given RESIDUAL HEAT REMOVAL System initial conditions, PREDICT how the system and/or plant parameters will respond to the manipulation of the following controls. .1 From Standby, Arming/Depressing Div 1(2) Manual Initiation Pushbutton. .2 From Standby, Arming/Depressing Div 1(2) Manual Containment Spray Initiation Pushbutton .3 System Operating in response to High Drywell Pressure, Arming/Depressing Div 1(2) Manual Containment Spray Initiation Pushbutton .4 System Operating in response to High Drywell Pressure, Operation of Pump Control Switch and/or F042A (B), (C) Control Switch

	Q58 219000 K3.01

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must determine how a LPCI initiation affects the ability of RHR to provide suppression pool temperature control.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the annunciator graphics in the stem and then determine how operation of RHR in SP Cooling Mode is affected to answer the question (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-1795659)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3312.01 Rev. 47 • CPS 5064.02 (2G) Rev. 32 • CPS 5065.02 (2D) Rev. 30a		
Training Objective	203000.15 Given RESIDUAL HEAT REMOVAL System initial conditions, PREDICT how the system and/or plant parameters will respond to the manipulation of the following controls. .3 System Operating in response to High Drywell Pressure		
Previous NRC Exam Use	ILT 12-1 NRC Exam		

59	ID: 2107268	Points: 1.00

The unit was operating at rated thermal power with:

- Division 1 DG out of service
- 4160V Bus 1A1 and 1B1 aligned as shown in the graphic below

At 1300, a transient occurred requiring the manual initiation of both containment spray subsystems.

Then, the following sequence of events occurred:

Time	Event
1303	Annunciators 5060-3D AND 5061-3D AC Undervoltage Second Level 4160V Bus was received on 4160V Bus 1A1 AND 1B1.
1306	RO resets and reinitiates the Division 1 and 2 Containment Spray subsystems.

Which of the following describes the expected status of Division 1 and 2 Residual Heat Removal (RHR) pumps?

A.	RHR Pump 1A and RHR Pump 1B are energized from the ERAT.

B.	RHR Pump 1A and RHR Pump 1B are energized from the RAT.
C.	RHR Pump 1A is de-energized. RHR Pump 1B is energized from the ERAT.
D.	RHR Pump 1A is de-energized. RHR Pump 1B is energized from DG 1B.

Answer:	D

Answer Justification / Plausibility Statements
D is correct: Under normal plant conditions with Div 1 and 2 DGs in standby, 2nd level undervoltage relays function as follows (reference 5060-3D & 5061-3D): After the Secondary Under Voltage Relay 15 sec time delay times, the Div 1 & 2 DG will start; the Div 1 & 2 Reserve (Main) feeder will open & the Div 1 & 2 Main (Reserve) feeder will lock-out; the 4.16KV Bus 1A1/1B1 (1AP07E/1AP09E) will be stripped of its loads; and the Div 1 & 2 DG will tie onto the bus. However, with Div 1 DG out of service and not available to re-energize 4160V Bus 1A1, RHR Pump 1A will be de-energized, and RHR Pump 1B will be energized from its respective DG (DG 1B). Per CPS 3312.01 caution before step 8.1.3.1, following a RHR Pump A(B) trip with a CNMT spray initiation signal present, the RHR pump breaker will NOT reclose on any further pump re-starts. RHR Pump Bkr is reset by <u>simultaneously</u> depressing the CNMT Spray A(B) Delay Timer Reset and CNMT Spray A(B) Seal-In Reset. Since this action was performed at 1306, RHR Pump 1B is energized from DG 1B and is operating in Containment Spray Mode. Incorrect Responses: A is incorrect but plausible. This is the expected response to a loss of bus voltage (1st Level Undervoltage) condition. B is incorrect but plausible and would be correct if the safety busses remained aligned to their normal sources until bus voltage degraded to the 1st level undervoltage setpoint. C is incorrect but plausible. The status of RHR Pump 1A is correct. RHR Pump 1B will be energized from its respective DG. This is the expected response to a loss of bus voltage (1st Level Undervoltage) condition.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
226001.K2.02	K2.02	2.9*	2.9*	2		5

System Name
RHR/LPCI: Containment Spray System Mode

Category Statement
Knowledge of electrical power supplies to the following: (CFR: 41.7)

K/A Statement
Pumps

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q59 226001 K2.02

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of electrical power supplies to the RHR Pumps that supply Containment Spray to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze conditions in the stem and then determine how the plant is impacted to answer the question (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 5060.03 (3D) Rev. 30b • 5061.03 (3D) Rev. 30c • CPS 3312.01 Rev. 47		
Training Objective	262001.03 DESCRIBE the function, operation, interlocks, trips, and power supplies of the following Auxiliary Power System components. .7 Divisional High Voltage Distribution System		
Previous NRC Exam Use	None		

60	ID: 2104996	Points: 1.00

The plant is in MODE 5.

The Fuel Pool Cooling and Cleanup Assist mode is in operation to maintain Spent Fuel Storage Pool (SFP) temperature.

Which of the following describes the associated system configuration?

A.	‘B’ RHR Pump is running and the FC Pumps are isolated.
B.	‘A’ RHR Pump is running and ‘A’ RHR Heat Exchanger is in service.
C.	ONE Suppression Pool Cleanup (SF) Pump is recirculating SFP water through BOTH FC Heat Exchangers.
D.	BOTH Suppression Pool Cleanup (SF) Pumps are recirculating SFP water through ONE FC Heat Exchanger.

Answer:	B

Answer Justification / Plausibility Statements
B is correct. Per CPS 3312.03 RHR - Shutdown Cooling (SDC) & Fuel Pool Cooling And Assist (FPC&A) step 6.10: The Residual Heat Removal (RH) System may be connected to the FC System to remove decay heat under the following conditions: • When the reactor is in a cold shutdown condition or is in the refueling mode, and • When an abnormal heat load has been produced in the pools and it appears that the pool water temperature will exceed 150°F. Per CPS 3312.03 section 8.2.2 Fuel Pool Cooling and Cleanup Assist, the following conditions must be met and alignments made to support the mode: • Step 8.2.2.1 - Verify plant is in Mode 4 or Mode 5 • Step 8.2.2.3 - Verify RHR Loop A in standby or available for use • Step 8.2.2.5 - FC system shutdown • Step 8.2.2.9.2 - 1E12-F066 RHR A Suct From Fuel Pool Cool Valve open • Step 8.2.2.11 - 1E12-F099 RHR A to Fuel Pool Cooling Valve open. • Step 8.2.2.13 - Start RHR Pump 'A' Water from the RHR 'A' pump discharge enters the FC system downstream of the FC Heat Exchangers through the manual RHR to FC Isolation Valve (1E12-F099) and flows to the Cask Storage Pool, Fuel Transfer Pool and Spent Fuel Storage Pool (SFP). Water from these pools flows into the FC surge tanks, out the FC Surge Tanks Outlet through the 1E12-F066 RHR A Suct From Fuel Pool Cool Valve to the RHR 'A' Pump and heat exchanger. Incorrect Responses: A is incorrect but plausible. This answer would be correct if the 'B' loop of RHR was tied to the Fuel Pool Cooling (FC) system. There is NO interconnection between RHR 'B' and the FC system. C and D are incorrect but plausible. These answers could be correct if the Suppression Pool Cleanup/Transfer (SF) system could be used as an alternate method to cool the Fuel Pool Cooling and Cleanup (FC) system. • The cross-tie between FC and SF can only be used to provide alternate Suppression Pool Cooling, NOT for supplemental SFP cooling. • Additionally, Alternate Suppression Pool Cooling uses both SF pumps and FPC&A uses one RHR pump.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
233000.K4.07	K4.07	2.7	2.9	2		9

System Name
Fuel Pool Cooling and Clean-up

Category Statement
Knowledge of FUEL POOL COOLING AND CLEAN-UP design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K/A Statement
Supplemental heat removal capability

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q60 233000 K4.07

233000.07	233000.07 Given the Fuel Pool Cooling & Cleanup system, DESCRIBE the systems supporting and the nature of the support.

233000.07	233000.07 Given the Fuel Pool Cooling & Cleanup system, DESCRIBE the systems supporting and the nature of the support.

CPS.722841	RO COMP #3

Other NRC Data

References Provided	None		
K/A Justification	Question meets the KA because the examinee must have knowledge of FC system design features which provide for supplemental heat removal capabilities (FPC&A) in order to answer this question.		
SRO-Only Justification	N/A		
Additional Information	Question is written at the memory level (low cog). The examinee has to recall the FPC&A lineup (1-P).		
NRC Exams Only			
Question Type	Bank (CL-ILT-637406)	**Difficulty**	N/A
Technical Reference and Revision #	CPS 3312.03 Rev. 11e		
Training Objective	233000.07 Given the Fuel Pool Cooling & Cleanup system, DESCRIBE the systems supporting and the nature of the support.		
Previous NRC Exam Use	None		

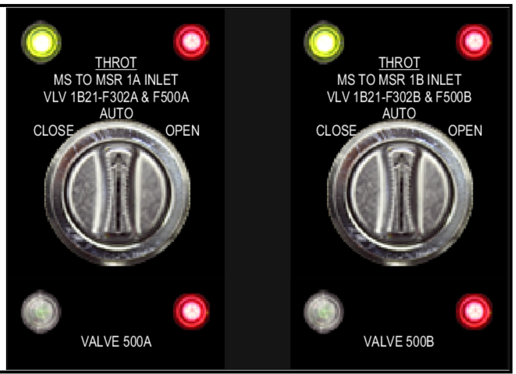
61	ID: 2106945	Points: 1.00

The CRS has directed you to place the Moisture Separator Reheaters (MSRs) in service.

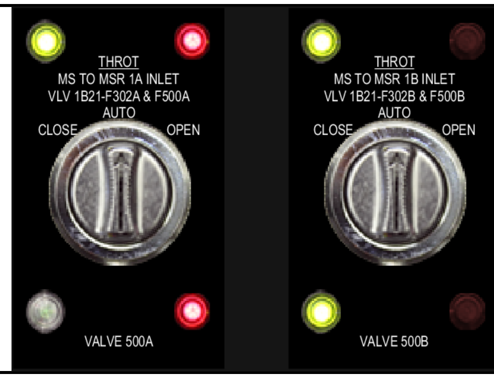
(1) The maximum heat up rate at the LP Turbine Inlets is ____(1)____ °F/hr.

(2) Which of the following graphics show how the 1B21-F302A/B Main Steam to Reheater Inlet Valves are required to be operated?

Case 1



Case 2



A.	(1) 25 (2) Case 1
B.	(1) 25 (2) Case 2
C.	(1) 75 (2) Case 1
D.	(1) 75 (2) Case 2

Answer:	C

Answer Justification / Plausibility Statements

C is correct.

Per CPS 3106.01 Moisture Separator Reheater, Limitations:

- Do not exceed ramp rates of 75°F/hr at the Low Pressure Turbine Inlets.
- Moisture Separator Reheaters (MSRs) shall be placed in and out of service as a pair.

Additionally, MSRs are operated as a pair to prevent uneven heating of LP Turbine casings.

Per BT10lr4 BWR Generic Fundamentals Chapter 10 Brittle Fracture And Vessel Thermal Stress, thermal stresses exist whenever temperature gradients are present in material.

Incorrect Responses:

A is incorrect but plausible. This answer would be correct if CPS 3106.01 Moisture Separator Reheater, Limitations limited the ramp rates for Low Pressure Turbine Inlets to 25°F/hr. The maximum side-to-side temperature differential at the LP turbine inlets should be limited to 25°F.

B is incorrect but plausible. This answer would be correct if:

- CPS 3106.01 Moisture Separator Reheater, Limitations limited the ramp rates for Low Pressure Turbine Inlets to 25°F/hr, and
- MSRs were manipulated one at a time such as when aligning Feedwater Heaters for service IAW CPS 3102.01 Extraction Steam/HTR Vent & Drains (ES, HD).

D is incorrect but plausible. This answer would be correct if MSRs were manipulated one at a time such as when aligning Feedwater Heaters for service IAW CPS 3102.01 Extraction Steam/HTR Vent & Drains (ES, HD).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
239001.K5.03	K5.03	2.7	2.9	2		3

System Name
Main and Reheat Steam System

Category Statement
Knowledge of the operational implications of the following concepts as they apply to MAIN AND REHEAT STEAM SYSTEM : (CFR: 41.5 / 45.3)

K/A Statement
Definition and causes of thermal stress

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q61 239001 K5.03

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must demonstrate knowledge of the operational implications of the causes of thermal stress as it applies to the Main And Reheat Steam system to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to analyze the graphics in the stem and then determine the expected operation of the MSR inlet valves based on the conditions presented in the stem (3-PEO).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3106.01 Rev. 25b • CPS 3102.01 Rev. 21d • BT10Ir4 Rev. 4		
Training Objective	239002.10 EXPLAIN the reasons for given MOISTURE SEPARATOR REHEATER System operating limits and precautions.		
Previous NRC Exam Use	None		

62	ID: 2104989	Points: 1.00

The plant was operating at 75% power with the Turbine Stop Valves (TSVs) and Turbine Control Valves (TCVs) positioned as follows:

- TSVs 1, 2, 3, and 4 - 100% (full open)
- TCVs 1, 2, and 3 - 32% open
- TCV 4 - 0% open (full closed)

THEN, TSV 1 failed closed.

3 minutes later, reactor power will be...

A.	75% with TCV #4 open.
B.	75% with TCV #4 closed.
C.	0% (scrammed) due to TSV/TCV position.
D.	0% (scrammed) due to high reactor pressure.

Answer:	B

Answer Justification / Plausibility Statements
B is correct. The Main Turbine is a tandem-compound unit with two double flow low-pressure sections. Steam flow from the Main Steam System is admitted to the high pressure turbine through four Turbine Stop Valves (TSV) and four steam flow Control Valves (CV), which are welded directly to the outlet of the Turbine Stop Valves. **An equalizing header connects the four TSV/CV pairings forming the Turbine Steam Chest**. Therefore, failure of TSV 1 will stop steam flow through TSV 1 only. The steam flowpath will be through TSVs 2, 3, and 4 and through TCVs 1, 2, and 3. The TCVs operate in a partial arc two-admission scheme. TCVs #1, 2, and 3 operate together and TCV #4 follows (TCV 4 will not open until TCVs 1-3 are full open). Since steam flow remains stable, TCVs 1-3 will remain at ~ 32% and TCV 4 will remain closed. Since reactor pressure will be stabilized, reactor power will remain at 75%. Incorrect Responses: A is incorrect but plausible. This answer would be correct if the flowpath through the TSVs and TCVs were paired (flow through each TSV and TCV pair only). Incorrect because the equalizing header connects the four TSV/CV pairings forming the Turbine Steam Chest. C is incorrect but plausible. This answer would be correct if a second TSV had failed closed. Per 5004-1D Div 1 or 4 TSV CL TRIP, a reactor scram will occur if annunciator 5004-3D DIV 1 OR 4 TCV FST CL & TSV TRIP BYP is NOT lit and any two instrument channels trip on Turbine Stop Valve closure. At 75% power, 5004-3D is reset. D is incorrect but plausible. This answer would be correct if the resulting RPV pressure spike was high enough to cause a Reactor Scram. Per CPS 5004-1C Div 1 or 4 Rx Vessel Hi Press Trip, the RPS trip setpoint is 1065 psig. Since RPV pressure is not significantly impacted with the reactor at 75% power, the high pressure scram setpoint will not be reached.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
241000.K6.11	K6.11	3.4	3.4	2		3

System Name
Reactor/Turbine Pressure Regulating System

Category Statement
Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM : (CFR: 41.7 / 45.7)

K/A Statement
Main stop/throttle valves

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q62 241000 K6.11

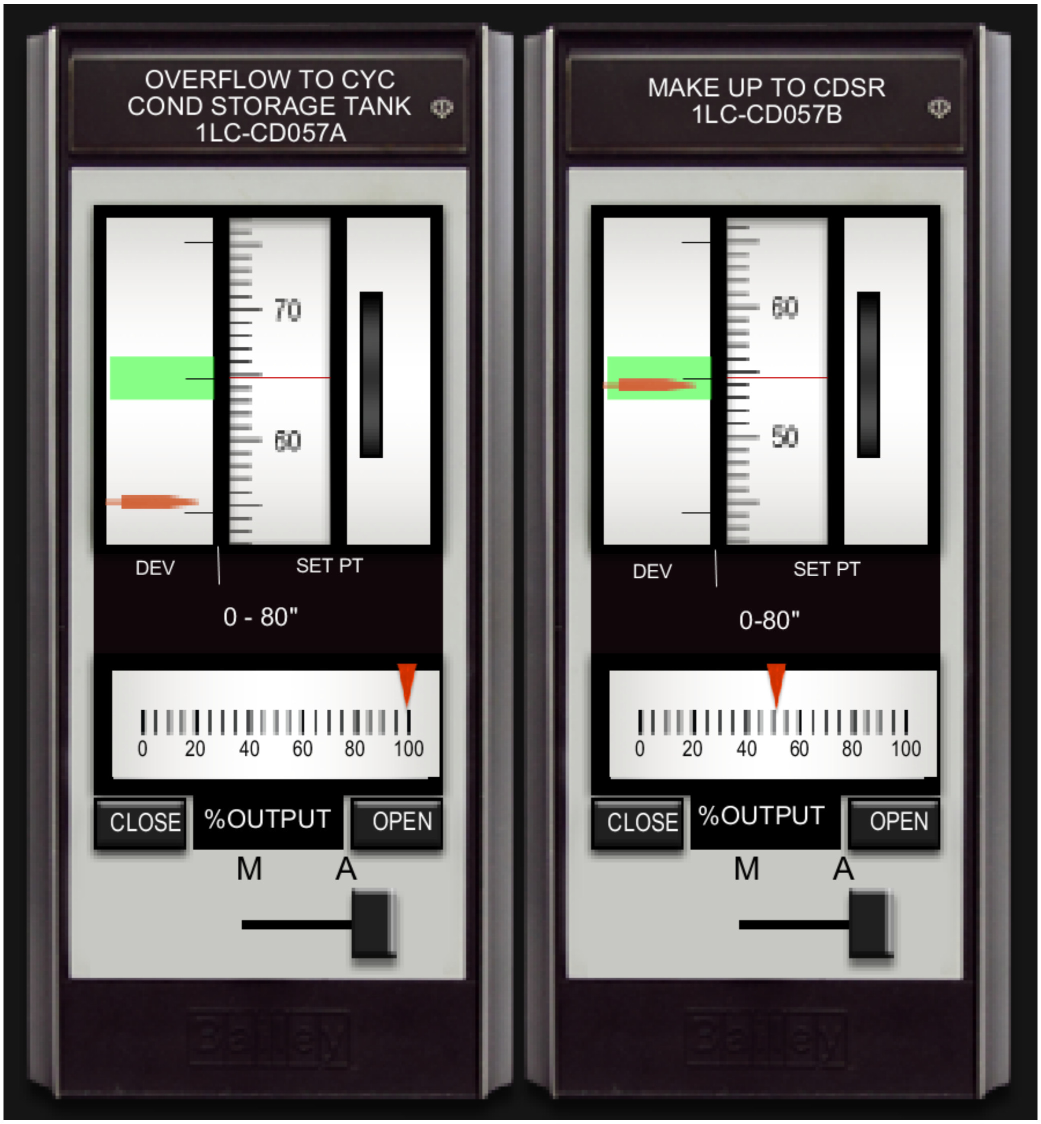
241000.03	241000.03 DESCRIBE the function, operation, interlocks, trips, and power supplies of the following ELECTROHYDRAULIC CONTROL System components. .1 Emergency Trip System .2 Steam Bypass Valves .3 Pressure Reducers .4 Bypass HPU Pumps .5 Hydraulic Power Unit Transfer Pump .6 Main EHC Fluid Pumps .7 EHC Transfer and Filter Pump .8 HPU Heaters and Fans .9 Coolers .10 Main Stop Valves .11 Control Valves .12 Combined Intermediate Valves .13 Fuller's Earth Filters .14 Permanent Magnet Generator (PMG)

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the candidate must demonstrate knowledge of the effect that a loss or malfunction of the TSVs will have on the Reactor/Turbine Pressure Regulating System. Specifically, the candidate must recognize the effect of a TSV closure on Reactor and Turbine Pressure Regulating System to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall the function of and the interlocks between the Main Turbine and the RPS system to answer the question (1-F/1-I)		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3305.01 Rev. 12b • CPS 5004.01 (1C & 1D) Rev. 28c • CPS 5004.03 (3D) Rev. 28b		
Training Objective	245000.09 DISCUSS the effect: A total loss or malfunction of the MAIN TURBINE (TG) System has on the plant.		
Previous NRC Exam Use	None		

63	ID: 2106962	Points: 1.00

The plant was operating at rated thermal power when the following indications were observed on 1H13-P680:



Which statement(s) is correct?

A.	Both controllers are operating normally.
B.	The Overflow Controller 1LC-CD057A is failed. The Makeup Controller 1LC-CD057B is operating normally.
C.	The Overflow Controller 1LC-CD057A is failed. The Makeup Controller 1LC-CD057B is <u>NOT</u> operating normally.
D.	The Makeup Controller 1LC-CD057B is failed. The Overflow Controller 1LC-CD057A is operating normally.

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Overflow Controller 1LC-CD057A is displayed with a 100% Output with Hotwell level below the controller green band. This indicates that the controller is failed. The Makeup Controller 1LC-CD057B is displayed with a 50% Output with Hotwell level being controlled in the controller green band. This indicates that the controller is responding normally to a low hotwell level by automatically opening the hotwell makeup valves to maintain hotwell level in the controller green band. Incorrect responses: A is incorrect but plausible. This answer would be correct if the overflow controller did not have a 100% demand (emergency overflow valve fully open). Hotwell level is normally maintained between 36-55 inches by the makeup and overflow controllers. Answer is plausible with the makeup controller in the green band. C is incorrect but plausible. This answer would be correct if the hotwell level was high which would cause the hotwell makeup valves to close and the % output on 1LC-CD057B would indicate 0%. D is incorrect but plausible. This answer would be correct if the makeup controller had failed at 50% output and the overflow controller was responding correctly to a high hotwell level. Incorrect because if 1LC-CD057B was failed at 50% open, the controller pointer (actual hotwell level) would be higher than the controller setpoint (green band), and the 1LC-CD057A controller pointer would be in the green band with a positive %output signal (> 0%).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
256000.A1.04	A1.04	2.9	2.9	2		2

System Name
Reactor Condensate System

Category Statement
Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CONDENSATE SYSTEM controls including: (CFR: 41.5 / 45.5)

K/A Statement
Hotwell level

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q35 256000 A3.06 V

	Q63 256000 A1.04

Other NRC Data

References Provided	None		
K/A Justification	Question meets the KA because the examinee must demonstrate the ability to monitor hotwell makeup and overflow which are parameters associated with operating the REACTOR CONDENSATE SYSTEM to select the correct response.		
SRO-Only Justification	N/A		
Additional Information	Question is written at the analysis/application level (high cog). The examinee must evaluate graphics presented in the stem and predict an event or outcome in order to select the correct response. (3-PEO)		
NRC Exams Only			
Question Type	Bank (CL-ILT-2027370)	**Difficulty**	N/A
Technical Reference and Revision #	CPS 3104.01 Rev. 32a		
Training Objective	256000.16 EVALUATE the following Condensate / Condensate Booster (CD / CB) indications/responses and DETERMINE if the indication/ response is expected and normal.		
Previous NRC Exam Use	None		

64	ID: 2107250	Points: 1.00

Which of the following describes the impact (if any) of a loss of DC MCC 1E on the Feedwater (FW) System?

A.	'A' TDRFP ONLY must be tripped locally if required.
B.	'A' TDRFP AND 'B' TDRFP must be tripped locally if required.
C.	The FW system is NOT directly affected by a loss of DC MCC 1E.
D.	If Feedwater Suction pressure begins to lower, Condensate Pump 'A' and Condensate Booster Pump 'A' must be started.

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Answer:	B

Answer Justification / Plausibility Statements
B is correct. Per CPS 4201.01 LOSS OF DC POWER, a loss of DC MCC 1E will cause a loss of tripping power to TDRFPs A & B. Trip TDRFPs locally, if required OR place TDRFP control on SLIM controller at minimum speed. Incorrect Responses: A is incorrect but plausible. This answer is partially correct ('A' TDRFP tripping power is lost). Incorrect because the 'B' TDRFP is similarly affected by the loss of DC MCC 1E. C is incorrect but plausible. With the plant at rated thermal power, a loss of DC MCC 1E will not impact the TDRFPs (both will continue to run in automatic without tripping power), so immediate mitigating actions are not required. The feedwater system, however, is afftected by a loss of DC MCC 1E (CD 'A' and 'C' / CB 'A' and 'C' minimum flow valves will fail open, and tripping power will be lost to TDRFPs 'A' and 'B'. D is incorrect but plausible. A loss of DC MCC 1E will result in a loss of power to the minimum flow valves for CD and CB Pumps 'A' and 'C', causing their respective minimum flow valves to fail open. Per CPS 4002.01 Abnormal RPV Level/Loss of Feedwater at Power, the immediate actions for feedwater malfunctions are to start/stop Condensate and Condensate Booster Pumps as necessary in support of level control. Incorrect because CD and CB 'A' Pumps do not have control power with DC MCC 1E deenergized.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
259001.A2.08	A2.08	2.5	2.6	2		2

System Name
Reactor Feedwater System

Category Statement
Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

K/A Statement
Loss of D.C. electrical power

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q64 259001 A2.08

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the candidate must demonstrate knowledge of the impact that a loss of the DC MCC 1E will have on the Reactor Feedwater system by selecting the correct mitigating action to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the comprehension level. The examinee has to recognize the interaction between DC Electrical Distribution system and Feedwater system, demonstrate an understanding of consequences due to a loss of DC MCC and correctly mitigate those effects (2-RI).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4002.01 Rev. 5c • CPS 4201.01 Rev. 8d • CPS 3103.01E001 Rev. 13a		
Training Objective	259001.07 Given the FEEDWATER system, DESCRIBE the systems supporting and the nature of the support. .8 DC Electrical Distribution System (DC)		
Previous NRC Exam Use	None		

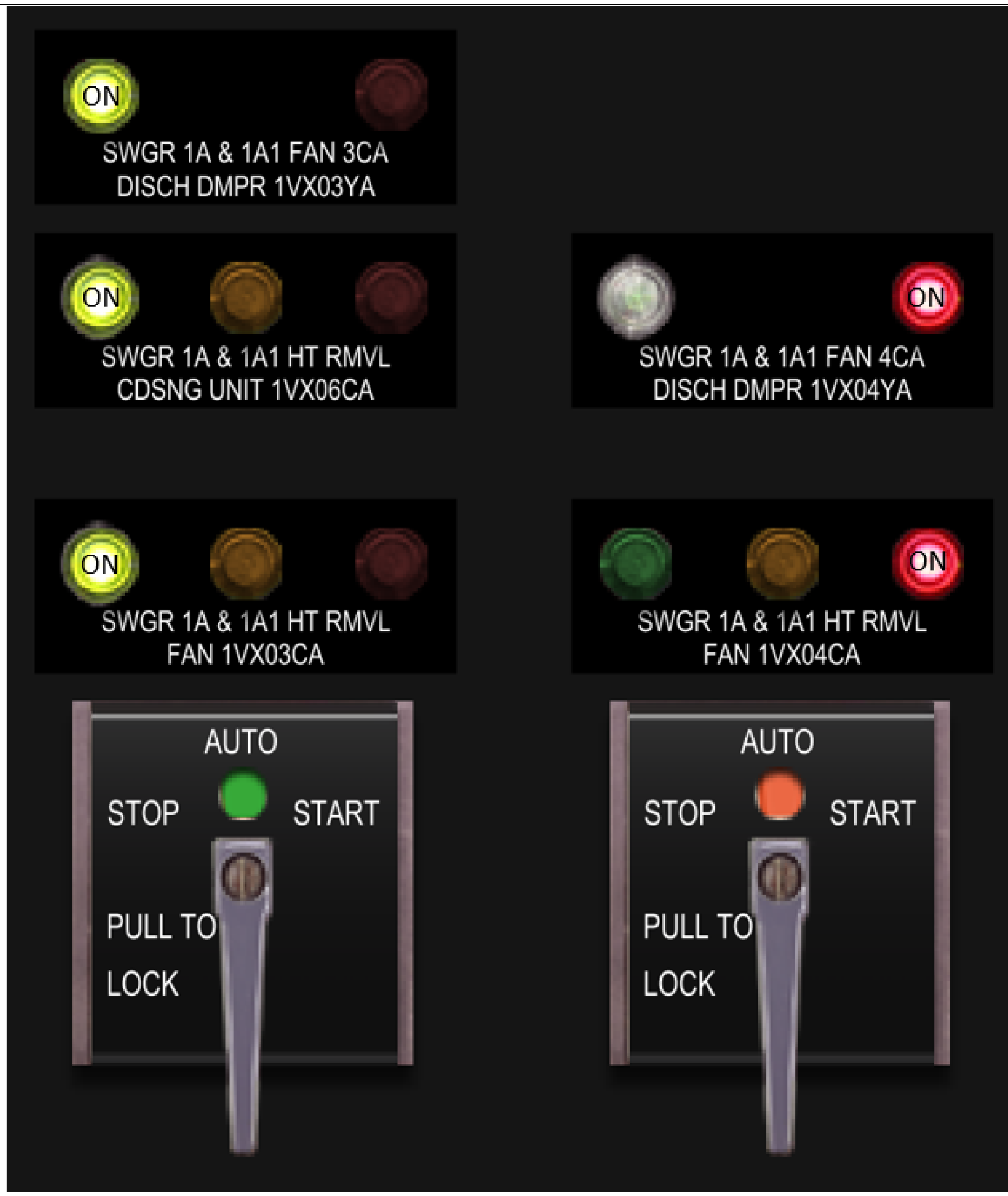
65	ID: 2107101	Points: 1.00

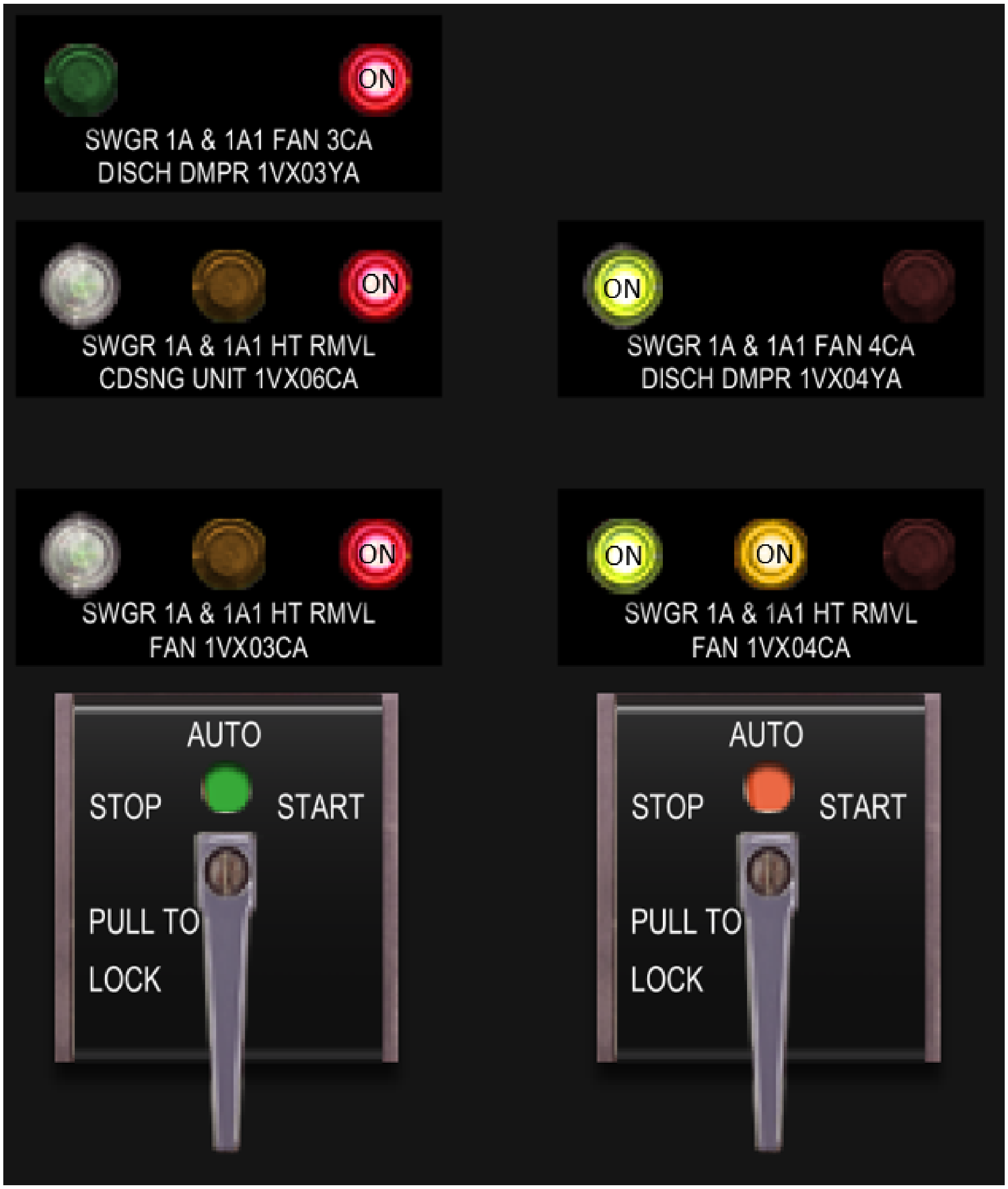
The plant was operating at rated thermal power.

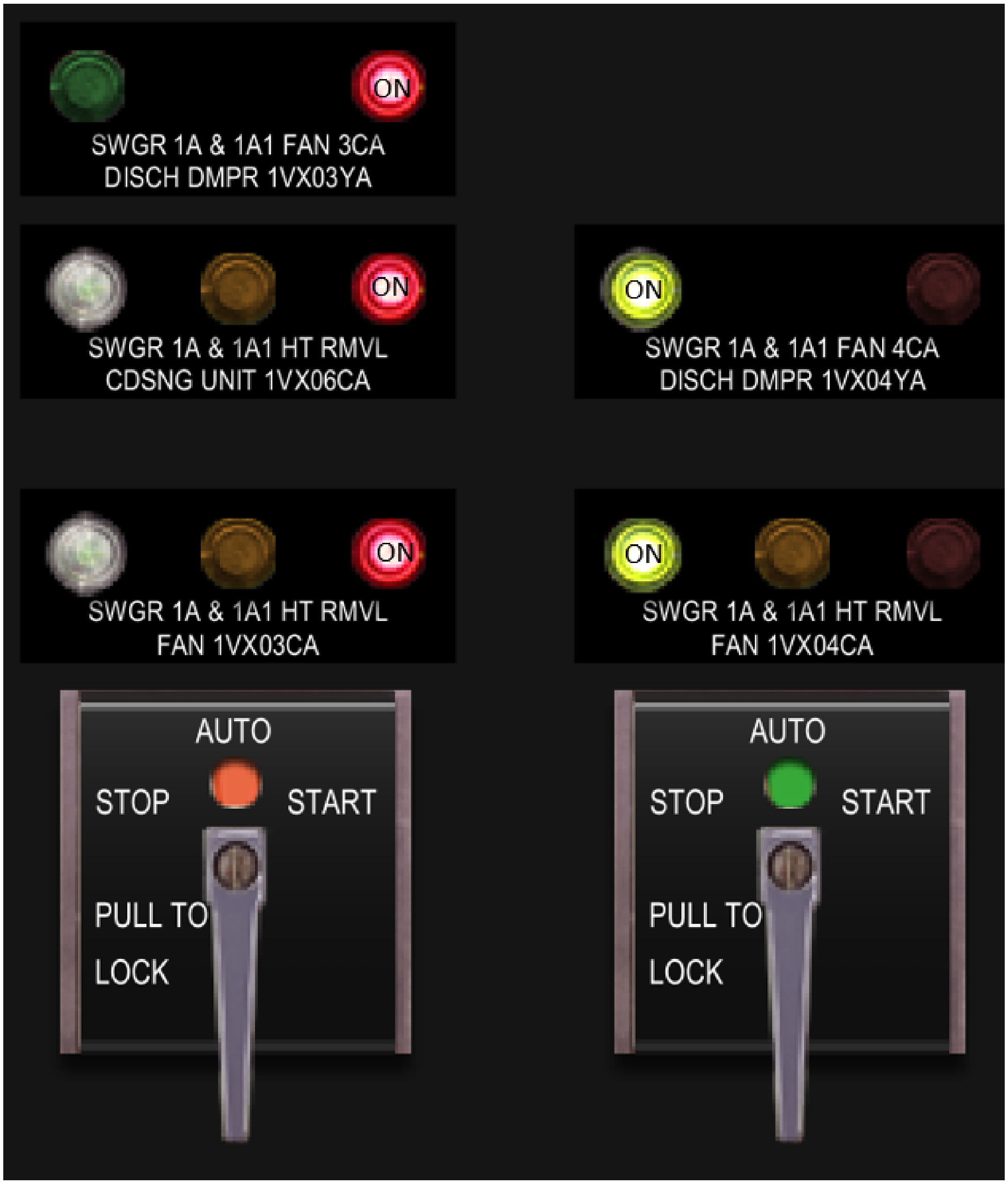
The Div 1 Essential Switchgear Heat Removal (VX) Fans 1VX03CA and 1VX04CA are in their normal configuration.

THEN, annunciator 5050-3D HIGH TEMP SWGR ROOM 1A1 was received.

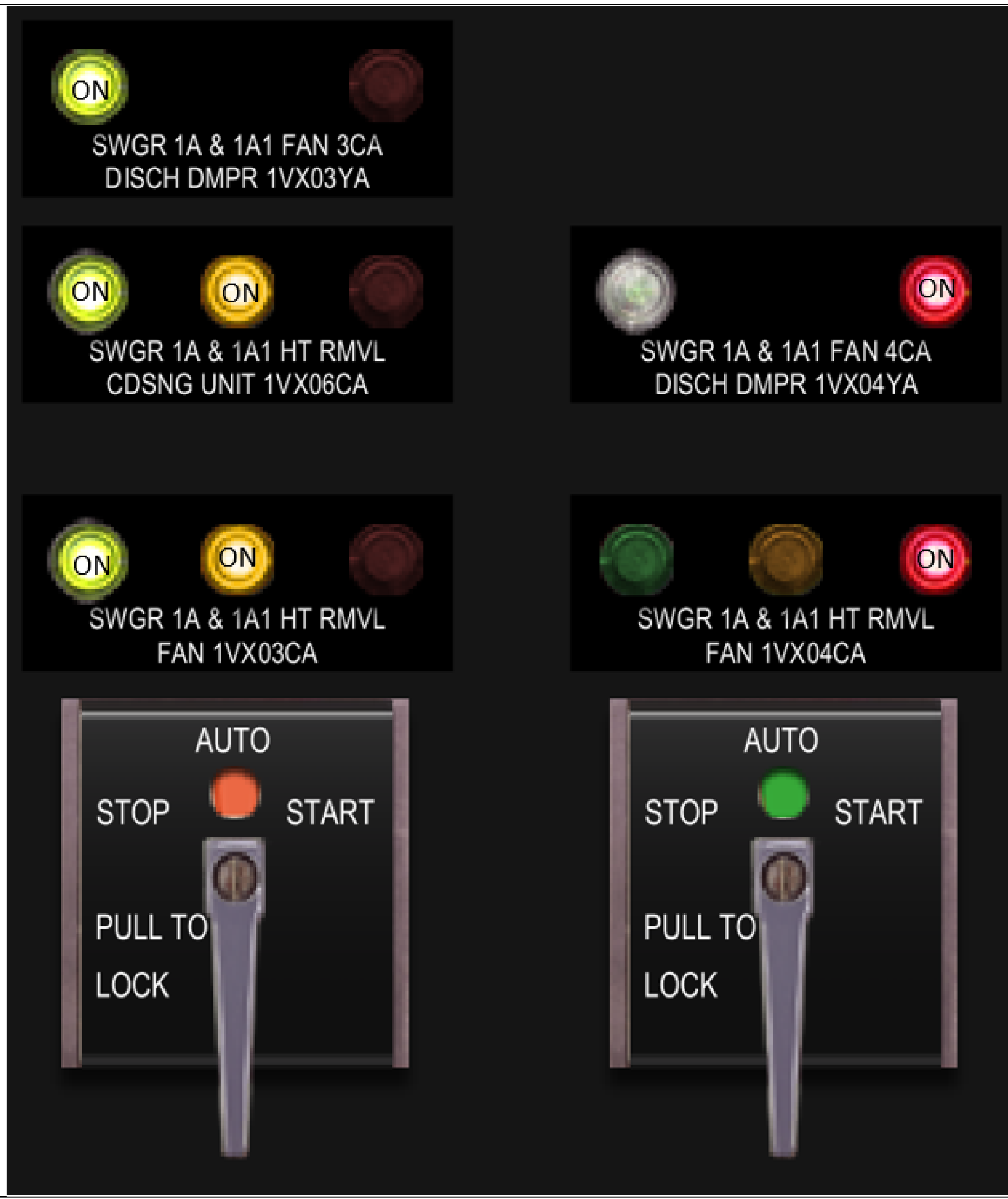
With no operator action, which graphic shows the expected alignment for Div 1 VX one minute later?

Alignment 1	ON SWGR 1A & 1A1 FAN 3CA DISCH DMPR 1VX03YA ON SWGR 1A & 1A1 HT RMVL CDSNG UNIT 1VX06CA ON SWGR 1A & 1A1 FAN 4CA DISCH DMPR 1VX04YA ON SWGR 1A & 1A1 HT RMVL FAN 1VX03CA ON SWGR 1A & 1A1 HT RMVL FAN 1VX04CA AUTO STOP START PULL TO LOCK AUTO STOP START PULL TO LOCK

Alignment 2	ON SWGR 1A & 1A1 FAN 3CA DISCH DMPR 1VX03YA ON SWGR 1A & 1A1 HT RMVL CDSNG UNIT 1VX06CA ON SWGR 1A & 1A1 FAN 4CA DISCH DMPR 1VX04YA ON SWGR 1A & 1A1 HT RMVL FAN 1VX03CA ON ON SWGR 1A & 1A1 HT RMVL FAN 1VX04CA AUTO STOP START PULL TO LOCK AUTO STOP START PULL TO LOCK

Alignment 3	

ON
SWGR 1A & 1A1 FAN 3CA
DISCH DMPR 1VX03YA
ON
SWGR 1A & 1A1 HT RMVL
CDSNG UNIT 1VX06CA
ON
SWGR 1A & 1A1 FAN 4CA
DISCH DMPR 1VX04YA
ON
SWGR 1A & 1A1 HT RMVL
FAN 1VX03CA
ON
SWGR 1A & 1A1 HT RMVL
FAN 1VX04CA
AUTO
STOP
START
PULL TO
LOCK
AUTO
STOP
START
PULL TO
LOCK

Alignment 4	

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

Answer:	B

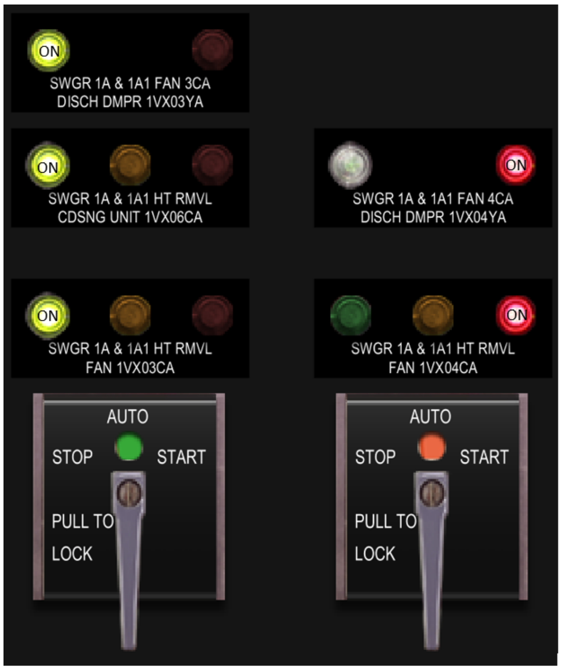
Answer Justification / Plausibility Statements

B is correct.

Per CPS 3412.01 Essential Switchgear Heat Removal (VX), section 8.1.1 System Startup, step 8.1.1.3 directs:

- starting 1VX04CA and verifying 1VX04YA opens
- establishing the standby feature of 1VX03CA by placing its control switch in AFTER STOP.

This alignment is depicted in the following graphic:



Per annunciator 5050-3D High Temp Swgr Room 1A1, a high temperature condition will cause:

- 1VX03CA to auto start
- 1VX03YA to open
- 1VX06CA to start
- 1VX04CA to trip
- 1VX04YA to close

This alignment is depicted in the alignment 2 graphic (B).

Incorrect Responses:

A is incorrect but plausible. This is the normal alignment for Div 1 VX and would be correct if the system did not respond to the conditions in the stem (stayed in its normal alignment).

C is incorrect but plausible if the system were normally aligned with 1VX03CA as the normal fan and 1VX04CA as the standby, and if the system did not respond to the conditions in the stem.

D is incorrect but plausible if the system were normally aligned with 1VX03CA as the normal fan and 1VX04CA as the standby, and the fans swapped on a high temperature condition.

EXAMINATION ANSWER KEY

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K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
288000.A3.01	A3.01	3.8	3.8	2		9

System Name
Plant Ventilation Systems

Category Statement
Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: (CFR: 41.7 / 45.7)

K/A Statement
Isolation/initiation signals

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q65 288000 A3.01

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to monitor the Switchgear Heat Removal System with a high temperature initiation signal present to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to choose a system alignment from 4 graphics based on analysis of conditions in the stem to answer the question (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3412.01 Rev. 16 • CPS 5050.03 (3D) Rev. 30c		
Training Objective	262003.03 DESCRIBE the function, operation, interlocks, trips, physical location, and power supplies of the following Switchgear Heat Removal System components. .1 Non-Safety Related Coil Cabinets .2 Non-Safety Related Supply Fans .3 Safety Related Supply Fans .4 Safety Related Condensing Units		
Previous NRC Exam Use	None		

66	ID: 2107038	Points: 1.00

While shifting drywell cooling IAW CPS 3320.01P001 Shifting Drywell Cooling Systems, the following sequence of events occurred:

Time	Event
1315	'B' DW chiller was started.
1330	'B' DW chiller tripped due to excessive loading rate.

The earliest time the 'B' DW chiller may be restarted is _____(1)_____.

An anti-recycle timer delays a chiller startup to prevent _____(2)_____.

A.	(1) 1335 (2) chiller tubes from freezeup
B.	(1) 1350 (2) chiller tubes from freezeup
C.	(1) 1335 (2) compressor motor from overheating
D.	(1) 1350 (2) compressor motor from overheating

Answer:	C

Answer Justification / Plausibility Statements

C is correct:

Per CPS 3320.01 Drywell Cooling (VP) and N-CL-OPS-288011 Drywell Cooling System, the drywell chillers are provided with an anti-cycle feature which prevents more than one **start attempt** in a 20 minute period. The 20-minute time delay allows the heat generated by the starting current to dissipate before another restart is attempted.

Incorrect Responses:

A is incorrect but plausible. This response would be correct if the anti-recycle timer protected the chiller tubes from freezeup. The chiller trip on low refrigerant temperature protects the cooler tubes from freezeup.

B is incorrect but plausible. This response would be correct if:

- the anti-recycle timer starts when the chiller trips (1330 + 20 min = 1350), and
- the anti-recycle timer protects the chiller tubes from freezeup. The chiller trip on low refrigerant temperature protects the cooler tubes from freezeup.

D is incorrect but plausible. This response would be correct if the anti-recycle timer starts when the chiller trips (1330 + 20 min = 1350).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.1.28	B2.1.28	4.1	4.1	3	N/A	N/A

System Name
Conduct of Operations

Category Statement
Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

EXAMINATION ANSWER KEY

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General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q66 2.1.28

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must have knowledge of the function and purpose of the anti-recycle timer which is a common feature on centrifugal refrigeration chillers (not just the DW Chillers) at CPS.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall information about interlocks to select the correct response (1-I).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3320.01 Rev. 22d • N-CL-OPS-288011 Rev. 4		
Training Objective	LP85801.2.1.28 Knowledge of the purpose and function of major system components and controls.		
Previous NRC Exam Use	None		

67	ID: 2107697	Points: 1.00

The plant is in Mode 5 performing core alterations.

Which of the following describes the correct configuration of the Reactor System Mode Switch?

A.	Refuel - key inserted
B.	Refuel - key removed
C.	Shutdown - key inserted
D.	Shutdown - key removed

Answer:	B

Answer Justification / Plausibility Statements

B is correct:

Per ORM 2.6.7 Reactor Mode Switch Position, the reactor mode switch shall be locked in Shutdown or Refuel position in Mode 5.

Per CPS 3007.01 PREPARATION AND RECOVERY FROM REFUELING OPERATIONS step 8.2.2, the Reactor Mode Switch is required to be locked in refuel when commencing core alterations.

Per CPS 1403.01 CPS KEY CONTROL PROGRAM step 8.1.12, any time the Reactor Mode Switch is required to be locked in a certain position per a Technical Specifications action item, the key shall be removed from the switch and placed in the Controlled Key Locker until the requirement no longer exists.

Incorrect Responses:

A is incorrect but plausible. During operations in Modes 1, 2, 3, and 4, the mode switch is not required to be locked, and the normal configuration is with the reactor mode switch key inserted.

C is incorrect but plausible.

- Part 1 - ORM 2.6.7 allows the reactor mode switch to be locked in shutdown or refuel in Mode 5. Core alterations are performed in Mode 5 but require the reactor mode switch to be in Refuel to enforce the refueling interlocks.
- Part 2 - During operations in Modes 1, 2, 3, and 4, the mode switch is not required to be locked, and the normal configuration is with the reactor mode switch key inserted.

D is incorrect but plausible - ORM 2.6.7 allows the reactor mode switch to be locked in shutdown or refuel in Mode 5. Core alterations are performed in Mode 5 but require the reactor mode switch to be in Refuel to enforce the refueling interlocks.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.1.31	B2.1.31	4.6	4.3	3	N/A	N/A

System Name
Conduct of Operations

Category Statement
Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. (CFR: 41.10 / 45.12)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q67 2.1.31

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee is required to demonstrate the ability to determine the correct configuration of the reactor mode switch when performing core alterations to answer the question.
SRO-Only Justification	N/A
Additional Information	This is a low cog question written at the memory level. The examinee has to recall procedure and ITS requirements for a switch configuration to answer the question (1-P).

NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• ORM 2.6.7 (page 10) - Rev. 80 • CPS 1403.01 Rev. 4b • CPS 3007.01 Rev. 21		
Training Objective	LP85801.2.1.31 Ability to locate control room switches / controls and indications and to determine that they are correctly reflecting the desired plant lineup.		
Previous NRC Exam Use	None		

68	ID: 2106965	Points: 1.00

The reactor was operating at 2431 MW thermal and 68.9 Mlbm/hr core flow.

THEN, core flow is raised to 80.5 Mlbm/hr.

The current core thermal power is approximately _________ MW Thermal.

A.	2682 - 2736
B.	2820 - 2876
C.	2992 - 3052
D.	3163 - 3227

Answer:	A

Answer Justification / Plausibility Statements
A is correct. Per CPS 3005.01 Unit Power Changes Figure 1: Stability Control & Power/Flow Operating Map, the original plant conditions indicate the reactor is operating on the 80% Flow Control Line (FCL). Because there is no rod movement, the candidate will follow the 80% FCL up to the new flow (80.5 Mlbm/hr). The intersection of 80.5 mlbh and the 80% FCL results in ~ 78% power. .78 * 3473 (RTP) = 2709 MWth. 2709 ± 1% = 2682 - 2736. Incorrect Responses: B is incorrect but plausible. This response would be correct if the original plant conditions had indicated the reactor was initially operating on the 85% FCL. The intersection of 80.5 mlbh and the 85% FCL results in ~ 82% power. .82 * 3473 = 2848 MWth. 2848 ± 1% = 2820 - 2876 C is incorrect but plausible. This response would be correct if the original plant conditions had incicated the reactor was initially operating on the 90% FCL. The intersection of 80.5 mlbh and the 90% FCL results in ~ 86% power. .87 * 3473 = 3022 MWth. 3022 ± 1% = 2992 - 3052 D is incorrect but plausible. This response would be correct if the original plant conditions had indicated the reactor was initially operating on the 95% FCL. The intersection of 80.5 mlbh and the 95% FCL results in ~ 92% power. .92 * 3473 = 3195 MWth. 3195 ± 1% = 3163 - 3227

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.1.25	B2.1.25	3.9	4.2	3	N/A	N/A

System Name
Conduct of Operations

Category Statement
Ability to interpret reference materials, such as graphs, curves, tables, etc. (CFR: 41.10 / 43.5 / 45.12)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q68 2.1.25

LP87437.01.03	LP87437.01.03 Given a Core Operating Map, core flow, reactor power and a specified change in power or flow; predict the final operating point on the map

Other NRC Data

References Provided	CPS 3005.01, Fig. 1		
K/A Justification	This question meets the KA because the examinee must demonstrate the ability to use reference material, such as a graph, to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the conditions presented in the stem and then using a power-flow map, predict the change in Rx power in order to answer the question (3-SPK/R).		
NRC Exams Only			
Question Type	Bank (CL-LC-1804)	**Difficulty**	N/A
Technical Reference and Revision #	CPS 3005.01 Rev. 43f		
Training Objective	LP87437.01.03 Given a Core Operating Map, core flow, reactor power and a specified change in power or flow; predict the final operating point on the map		
Previous NRC Exam Use	None		

69	ID: 2106966	Points: 1.00

A manual throttle valve will be positioned CLOSED three (3) turns from FULL OPEN for a surveillance test.

Which of the following is procedurally required for this manipulation?

A.	First Check
B.	Peer Check
C.	Concurrent Verification
D.	Independent Verification

Answer:	C

Answer Justification / Plausibility Statements

C is correct

Per HU-AA-101 Human Performance Tools and Verification Practices, CONCURRENT VERIFICATION is defined as:

The act of two qualified individuals verifying the correct component identification and performing subsequent component manipulation, which, if performed incorrectly, would cause an irrecoverable condition with immediate adverse consequences to plant operation.

Per HU-AA-101 Human Performance Tools and Verification Practices, Section 4.3.4 Concurrent Verifications are performed for the following activities:

Concurrent Verification (CV) may be applicable to -but not limited to- component manipulations, clearance application / removal, performance of a procedure or other activities that removes equipment from service. Examples are:

- Fuse removal and replacement
- Lifting and re-landing leads
- Booting relay contacts
- Jumper installation
- **Valve throttling**
- Breaker manipulation
- Switch manipulation
- Gagging of valves

Incorrect Responses:

A is incorrect but plausible because FIRST CHECK is performed as a configuration control method by operators when manipulating equipment to ensure the component being manipulated is performed on the proper unit and train. It could be used when manipulating a throttle valve, but is not adequate to ensure proper throttle valve positioning without the use of concurrent verification.

B is incorrect but plausible because PEER CHECK is performed as a configuration control method by operators when manipulating equipment. Peer check is not adequate when positioning throttle valves, however, because the independent action of verifying the correct component identification is missing when performing peer checks.

D is incorrect but plausible because INDEPENDENT VERIFICATION is performed as a configuration control method by operators when manipulating equipment. Independent verification is not adequate when positioning throttle valves, however, because the correct positioning cannot be verified independently.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.2.14	B2.2.14	3.9	4.3	3	N/A	N/A

System Name
Equipment Control

Category Statement
Knowledge of the process for controlling equipment configuration or status. (CFR: 41.10 / 43.3 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.3	43.3

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q69 2.2.14

LP85801.2.1.29	LP85801.2.1.29 Knowledge of how to conduct and verify valve / equipment lineups.

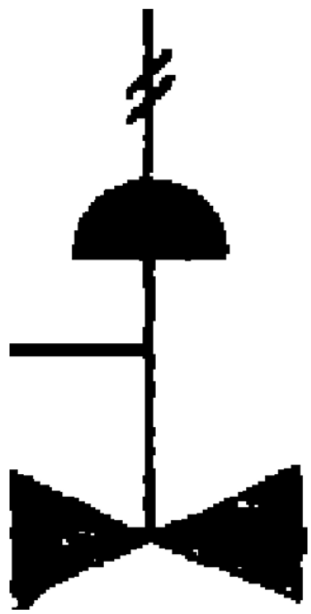
Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must determine the correct method for controlling equipment configuration when performing lineups on manual throttle valves.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall procedure steps to select the correct response (1-P).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A11074)	**Difficulty**	N/A
Technical Reference and Revision #	HU-AA-101 Rev. 10		
Training Objective	LP85802.2.2.14 Knowledge of the process for controlling equipment configuration or status.		
Previous NRC Exam Use	None		

70	ID: 2115376	Points: 1.00

During the review of a mechanical drawing, you note the schematic representation below.

The valve represented is normally _____(1)_____ at rated thermal power and fails _____(2)_____ on a loss of air.



A.	(1) open (2) open

B.	(1) open (2) closed

C.	(1) closed (2) open

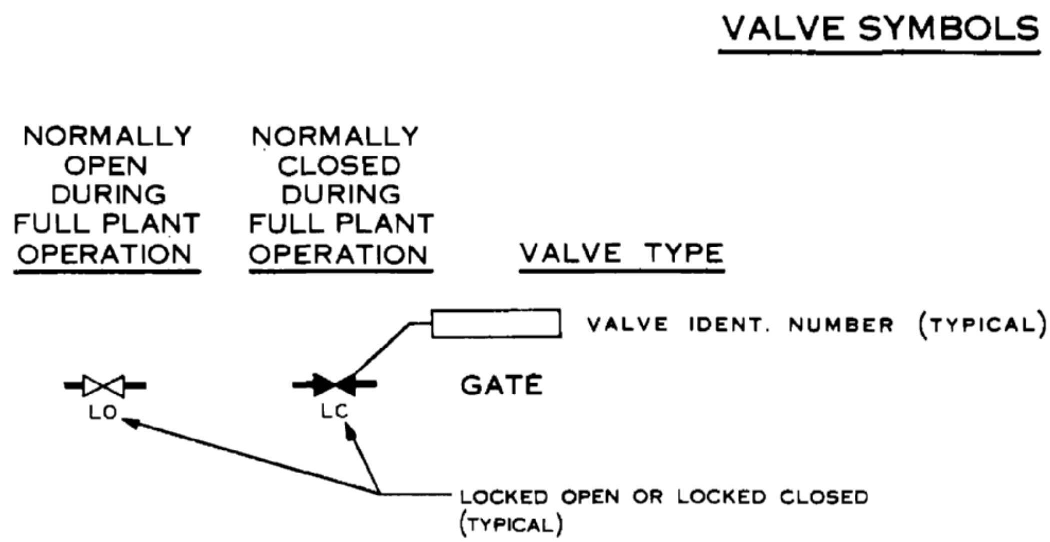
D.	(1) closed (2) closed

Answer:	C

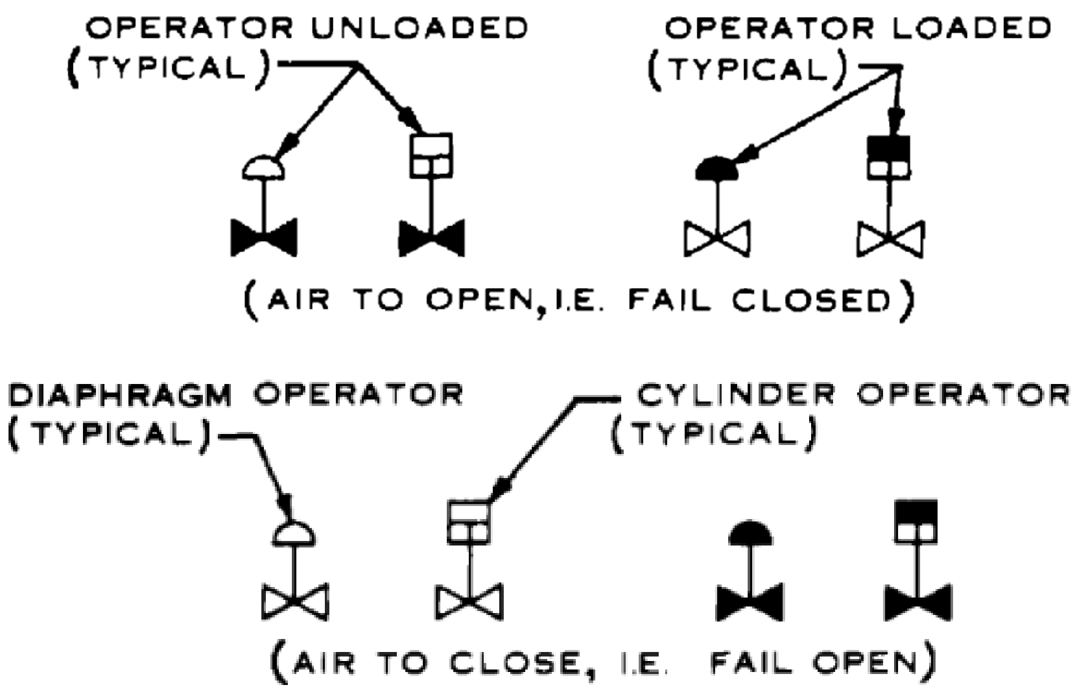
Answer Justification / Plausibility Statements

C is correct:

Per M05-1001 Sht. 001 PID Standard Symbols, valves on M05 drawings are depicted in their normal full plant operation position. A valve that is clear is normally open during full plant operation and a valve that is blackened in is normally closed at full plant operation.



In addition, air operated valves are shown in their normal position at full plant operation with the state of their air operators (loaded or unloaded). If the valve symbol shows the valve normally closed with its air operator loaded the valve fails open. If the valve symbol shows the valve normally open with its air operator loaded the valves fails closed.



Therefore, in the graphic shown in the question stem, the valve is normally shut during full plant operation and fails open on a loss of air.

Incorrect Responses:

A is incorrect but plausible. This answer is partially correct and would be completely correct if the symbology for a normally open air operated valve (at full plant operation) was indicated as blackened in.

B is incorrect but plausible. This answer would be correct if:

- the symbology for a normally open air operated valve (at full plant operation) was indicated as blackened in, and
- the blackened in dome was indicative of an unloaded air operator.

D is incorrect but plausible. This answer is partially correct and would be completely correct if the blackened in dome was indicative of an unloaded air operator.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.2.41	B2.2.41	3.5	3.9	3	N/A	N/A

System Name

Category Statement
Ability to obtain and interpret station electrical and mechanical drawings. (CFR: 41.10 / 45.12 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q70 2.2.41

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must be able to interpret a symbol found on a mechanical print in order to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall facts pertaining to mechanical drawing symbology in order to select the correct response (1-F).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	M05-1001 SH001 Rev. D		
Training Objective	LP85802.2.2.41 Ability to obtain and interpret station electrical and mechanical drawings. (Moved from 2.1.24)		
Previous NRC Exam Use	None		

71	ID: 2104928	Points: 1.00

You are an operator currently performing a task in a radiologically controlled area.

The RWP controlling your task includes the following information:

- ED Dose Alarm 240 mrem
- Dose Rate Alarm 800 mrem/hr

The actual dose rate in your work area is 60 mrem/hr.

You have accumulated 191 mrem and require more time in the area to complete the task.

Which of the following:

- is permitted by RP-AA-403 Administration of the Radiation Work Permit Program, and
- maximizes your time working on the task?

A.	Exit the area immediately.
B.	Continue to work in the area for 25 minutes.
C.	Continue to work in the area for 48 minutes.
D.	Continue to work in the area until the ED Dose Alarm is received.

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per RP-AA-403 Administration of the Radiation Work Permit Program Attachment 2 Manual RWP, back out criteria is reached when accumulated dose is 80% of the dose alarm. 191 mrem is 79.6% of the ED dose alarm setpoint (191 / 240 = 79.6%) and the operator should exit the area immediately. Incorrect Responses: B is incorrect but plausible. This answer would be correct if the radiation worker were allowed to continue working until the dosimetry reaches 90% of the ED Dose Alarm. If the worker stays in the area for 25 minutes, his/her cumulative dose will be 216 mrem, or 90% of the ED Dose Alarm Setpoint. This value exceeds the 80% of ED Dose Alarm limit (192 mrem) in RP-AA-1008. C is incorrect but plausible. This answer would be correct if the radiation worker were allowed to continue working until his/her cumulative dose approaches the accumulated dose alarm setpoint. If the worker stays in the area for 48 minutes, his/her cumulative dose will be 239 mrem, or 99.6% of the ED Dose Alarm setpoint. This value exceeds the 80% of ED Dose Alarm limit (192 mrem) in RP-AA-1008. D is incorrect but plausible. This answer would be correct if the radiation worker were allowed to continue working until the ED Dose Alarm <u>is received</u>. If the worker stays in the area until the ED Dose Alarm is received, his/her cumulative dose will be 240 mrem, or 100% of the ED Dose Alarm setpoint. This value exceeds the 80% of ED Dose Alarm limit (192 mrem) in RP-AA-1008.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.3.07	B2.3.07	3.5	3.6	3	N/A	N/A

System Name
Radiation Control

Category Statement
Ability to comply with radiation work permit requirements during normal or abnormal conditions. (CFR: 41.12 / 45.10)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.12	41.12

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q71 2.3.7

LP85803.2.3.7	LP85803.2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must be able to determine when the RCA must be exited based on knowledge of back out criteria contained in RP-AA-403 Administration of the Radiation Work Permit Program to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This question is a high cog question written at the analysis and application level. The examinee must solve a problem using knowledge of the ALARA principle (time, distance and shielding), RWP setpoints and applying knowledge of RWP requirements to answer the question (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-N12072)	**Difficulty**	N/A
Technical Reference and Revision #	RP-AA-403 Rev. 10		
Training Objective	LP85803.2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions.		
Previous NRC Exam Use	CPS ILT 12-1 NRC Exam		

72	ID: 2104924	Points: 1.00

The HIGHEST power level that will still allow a Drywell entry is…

A.	2%
B.	4%
C.	8%
D.	10%

Answer:	B

Answer Justification / Plausibility Statements
B is correct. Per RP-CL-460-1002 Drywell Entries, step 5.2.3, drywell (DW) entries are not allowed in Mode 1. DW entries during a controlled shutdown are allowed in Mode 2 with Reactor Power equal to or less than 5% Rx power. DW entries during plant startup are allowed up to, but not greater than 5% Rx power. It is expected that Operations will continue to maintain Rx power stable or decreasing (during shutdown) to meet this requirement. Incorrect Responses: A is incorrect but plausible. At 2% reactor power, the reactor is in Mode 2 and drywell radiation levels would be acceptable for drywell entry. However, 2% is not the HIGHEST power level that would still allow a drywell entry. C and D are incorrect but plausible. 8% to 10% are the power levels specified in CPS 3002.01 Heatup and Pressurization, for transition into Mode 1. Although drywell radiation levels would be relatively low, RP-CL-460-1002 Drywell Entries specifies that DW entries are not allowed in Mode 1 or with reactor power above 5%.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.3.12	B2.3.12	3.2	3.7	3	N/A	N/A

System Name

Category Statement
Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.12	41.12

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q72 2.3.12

LP85803.2.3.12	LP85803.2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because it requires knowledge of radiological safety principles (ALARA - maximum power level requirement) pertaining to licensed operator duties (DW entries).		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall procedure steps/precautions to select the correct response (1-P).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A12071)	**Difficulty**	N/A
Technical Reference and Revision #	• RP-CL-460-1002 Rev. 2 • CPS 3002.01 Rev. 32f		
Training Objective	LP85803.2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.		
Previous NRC Exam Use	None		

73	ID: 2115377	Points: 1.00

A Site Area Emergency has been declared.

The Technical Support Center (TSC) has been activated and has assumed command and control.

Who has Command Authority?

A.	TSC Director
B.	Operations Manager
C.	Station Emergency Director
D.	Corporate Emergency Director

Answer:	C

Answer Justification / Plausibility Statements
C is correct: Per EP-AA-112-200 TSC Activation and Operation, the Station Emergency Director supervises and directs the station emergency response organization. The Station Emergency Director's responsibilities include organizing and coordinating onsite emergency efforts. Additionally, the Station Emergency Director has requisite authority, plant operating experience and qualifications to implement inplant recovery operations. Incorrect Responses A is incorrect but plausible. The TSC Director is an actual TSC ERO position, but the TSC Director reports to the Station Emergency Director and is responsible for the content of information transmitted from the TSC to other facilities or agencies and for supporting overall TSC activities. B is incorrect but plausible. The Operations Manager is an actual TSC ERO position, but the Operations Manager reports to the Station Emergency Director and determines the extent of station emergencies, initiates corrective actions, and implements protective actions for onsite personnel. In the event that the Station Emergency Director becomes incapacitated and can no longer fulfill the designated responsibilities, the Operations Manager will assume the responsibilities of the Station Emergency Director until relieved by another qualified Station Emergency Director. D is incorrect but plausible. The Corporate Emergency Director is an actual EOF ERO position, The Corporate Emergency Director is the designated individual who has the authority, management ability, and technical knowledge to manage Exelon Nuclear's Emergency Response activities in the Emergency Operations Facility (EOF). The EOF shall achieve Minimum Staffing and facility activation within 60 minutes of an Alert or higher declaration.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.4.37	B2.4.37	3.0	4.1	3	N/A	N/A

System Name
Emergency Procedures /Plan

Category Statement
Knowledge of the lines of authority during implementation of the emergency plan. (CFR: 41.10 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q73 2.4.37

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee is required to demonstrate knowledge of the lines of authority during implementation of the emergency plan to answer this question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall facts from a procedure to answer the question (1-F).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A12075)	**Difficulty**	N/A
Technical Reference and Revision #	• EP-AA-112-200 Rev. 12 • EP-AA-112-400 Rev. 14		
Training Objective	LP85804.2.4.37 Knowledge of the lines of authority during implementation of the emergency plan.		
Previous NRC Exam Use	None		

74	ID: 2107002	Points: 1.00

The plant was operating at rated thermal power.

THEN, a pressure transient occurred requiring entry into EOP-1 RPV Control.

Actions are in progress in the RPV pressure and level legs.

THEN, RPV level drops to 0".

How is EOP-1 executed?

A.	Reentry at the top of EOP-1 RPV Control is required.
B.	ONLY reentry at the start of the RPV level leg is required.
C.	Continue execution of EOP-1 RPV Control per the EOP flow chart.
D.	Reentry at the start of BOTH the RPV pressure and level legs is required.

Answer:	A

Answer Justification / Plausibility Statements
A is correct. Per CPS 1005.09 Emergency Operating Procedure (EOP) And Severe Accident Guideline (SAG) Program, Section 8.12.3 EOP Entry / Reentry: .3 An EOP shall be reentered upon each receipt of an entry condition. Incorrect Responses: B is incorrect but plausible. This answer would be correct if reentry into the affected leg (RPV Water Level) of EOP-1 was required, similar to the rounded rectangle labeled "Return from EOP-3" which serves as a reentry point in EOP-1A ATWS RPV Control. C is incorrect but plausible. This answer would be correct if the specific guidance in OP-CL-101-111-1001 Strategies For Successful Transient Mitigation on EOP Execution (i.e. All steps must be executed in their specified order when executing control legs of the EOP's) took precedence over the guidance of CPS 1005.09. D is incorrect but plausible. This answer would be correct if reentry into all legs of EOP-1 were required, similar to the rounded rectangle labeled "Return from EOP-3" which serves as a reentry point in EOP-1A ATWS RPV Control.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.4.14	B2.4.14	3.8	4.5	3	N/A	N/A

System Name
Emergency Procedures /Plan

Category Statement
Knowledge of general guidelines for EOP usage. (CFR: 41.10 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

EXAMINATION ANSWER KEY

ILT 18-1 NRC RO Written Exam

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

LP87551.01.03	LP87551.01.03 Explain the principles associated with the selection/specification of EOP entry conditions.

	Q74 2.4.14

Other NRC Data

References Provided	None		
K/A Justification	Question meets the KA because the examinee has to demonstrate general guidelines for EOP usage (EOP Reentry) during execution of EOP-1 to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the conditions presented in the stem and determine how an EOP is to be executed in order to answer the question (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-LC-1055)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 1005.09, Rev. 11 • OP-CL-101-111-1001, Rev. 15b		
Training Objective	LP85804.2.4.14 Knowledge of general guidelines for EOP usage.		
Previous NRC Exam Use	None		

75	ID: 2107247	Points: 1.00

A plant event has occurred requiring the MCR to be evacuated.

Per CPS 4003.01 Remote Shutdown (RS), the ____(1)____ Reactor Operator (RO) will report to the Remote Shutdown Panel (RSP) and the ____(2)____ will proceed to the Division 1 Diesel Generator Room.

A.	(1) 'A' (2) 'B' RO
B.	(1) 'B' (2) 'A' RO
C.	(1) 'A' (2) Safe Shutdown Operator (SSO)
D.	(1) 'B' (2) Safe Shutdown Operator (SSO)

Answer:	C

Answer Justification / Plausibility Statements
C is correct. Per CPS 4003.01 Remote Shutdown (RS): • The 'A' RO reports to the RSP and establishes plant control per 4003.01H003 Section 1.0, RSP - Remote Area Duties. • Safe Shutdown Operator (SSO) to proceed to the Division 1 (Div 1) Diesel Generator (DG) Room. Incorrect Responses: A is incorrect but plausible. This response would be correct if the 'B' RO was directed to report to the Div 1 DG Room in accordance with CPS 4003.01. The SSO is the operator designated by CCPS 4003.01 to report to the Div 1 DG Room. Additionally, per OP-CL-101-102-1001 CPS Minimum On-Shift Staffing Functions, The SSO can not be the 'B' RO. B is incorrect but plausible. This response would be correct if: • the 'B' RO was directed to report to the RSP in accordance with CPS 4003.01. The 'B' RO performs CPS 4003.01H002 MCR - 'B' CRO Hard Card as part of the Initial MCR Actions prior to Evacuation (CPS 4003.01), but does not have any other position designated responsibilities. AND • the 'A' RO was directed to report to the Div 1 DG Room in accordance with CPS 4003.01. The 'A' RO reports to the RSP and establishes plant control per 4003.01H003 Section 1.0, RSP - Remote Area Duties. Additionally, per OP-CL-101-102-1001 CPS Minimum On-Shift Staffing Functions, The SSO can not be the 'A' RO. D is incorrect but plausible. This response would be correct if the 'B' RO was directed to report to the RSP in accordance with CPS 4003.01. The 'B' RO performs CPS 4003.01H002 MCR - 'B' CRO Hard Card as part of the Initial MCR Actions prior to Evacuation (CPS 4003.01), but does not have any other position designated responsibilities.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.4.12	B2.4.12	4.0	4.3	3	N/A	N/A

System Name
Emergency Procedures /Plan

Category Statement
Knowledge of general operating crew responsibilities during emergency operations. (CFR: 41.10 / 45.12)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

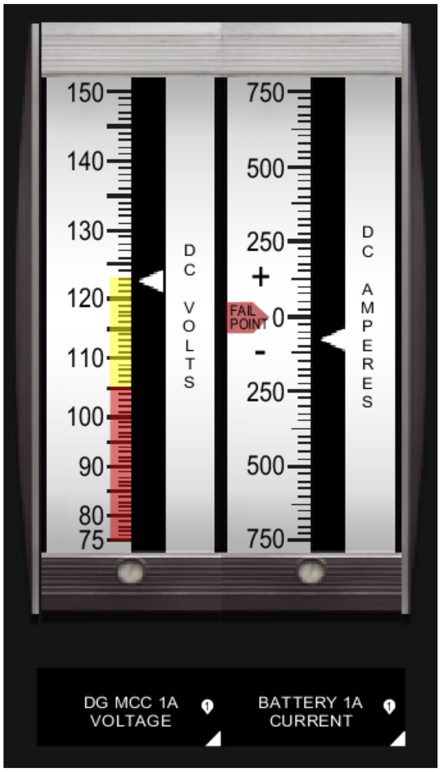
	Q75 2.4.12

Other NRC Data

References Provided	None		
K/A Justification	Question meets the KA because the examinee has to demonstrate knowledge of operating crew responsibilities during emergency operations (MCR evacuation) to answer the question.		
SRO-Only Justification	N/A		
Additional Information	This is a low cog question written at the memory level. The examinee must recall ITS facts to select the correct response (1-F).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4003.01, Rev. 18a • OP-CL-101-102-1001, Rev. 7d		
Training Objective	LP85804.2.4.12 Knowledge of general operating crew responsibilities during emergency operations.		
Previous NRC Exam Use	None		

1	ID: 1097362	Points: 1.00

The unit was operating at rated thermal power with NO LCOs in effect when the following indications were observed on 1H13-P877-5060:



The following are excerpts from ITS 3.8.4 DC Sources - Operating, and ITS 3.8.1 AC Sources - Operating:

3.8.4 DC Sources - Operating		
Condition	Required Action	Completion Time
A. One battery charger on Division 1 or 2 inoperable.		
B. One battery on Division 1 or 2 inoperable.		

3.8.1 AC Sources - Operating		
Condition	Required Action	Completion Time
B. One required DG inoperable.		

Which ITS LCO Condition(s) must be entered?

A. 3.8.4 Condition A ONLY

B. 3.8.4 Condition B ONLY

C. 3.8.4 Condition A AND 3.8.1 Condition B ONLY

D. 3.8.4 Condition B AND 3.8.1 Condition B ONLY

Answer: A

Answer Justification / Plausibility Statements
A is correct: The DC MCC 1A Voltage and Battery 1A Current indications provided in the stem indicate that AC power has been lost to Battery Charger 1A requiring entry into ITS 3.8.4 Condition A. Per OP-CL-108-104-1001 ITS LCO/ORM OR/ODCM OR EVALUATIONS AND GUIDANCE FOR SAFETY FUNCTION DETERMINATION section 2.10, when the DC Sources LCO is not met but the DC Distribution System is energized either from its associated battery or battery charger, the supported systems should not be declared inoperable, except as provided for in the DC Sources ACTIONS. This only applies to the Div 3 or 4 DC electrical power subsystems, which requires declaring High Pressure Core Spray inoperable (3.8.4 Condition E). Incorrect Responses: B is incorrect but plausible based on the degraded battery voltage indication in the stem. Battery operability is based on the following: • parameter values listed in ITS LCO 3.8.6 Battery Parameters. These values are based on float values with the associated charger in service, which is not the case in this question. • minimum design voltage (ITS B3.8.4 page B3.8-50) of 105VDC. Battery voltage in the stem is above this value. C is incorrect but plausible. Div 1 DC is a support system for the Div 1 DG, however ITS LCO 3.0.6 does not require cascading support-supported LCOs unless a loss of safety function exists. With Div 2 DG operable, a loss of safety function does not exist, therefore ITS 3.8.1 Condition B is not required to be entered. D is incorrect but plausible: Battery operability is based on the following: • parameter values listed in ITS LCO 3.8.6 Battery Parameters. These values are based on float values with the associated charger in service, which is not the case in this question. • minimum design voltage (ITS B3.8.4 page B3.8-50) of 105VDC. Battery voltage in the stem is above this value. and Div 1 DC is a support system for the Div 1 DG, however OP-CL-108-104-1001 does not require cascading support-supported LCOs unless a loss of safety function exists. With Div 2 DG operable, a loss of safety function does not exist, therefore ITS 3.8.1 Condition B is not required to be entered.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295003.AA2.03	AA2.03	3.2	3.5	1		6

System Name
Partial or Complete Loss of A.C. Power

Category Statement
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
Battery status: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q1/76 295003 A2.03

263000.12	263000.12 Given BATTERY & DC DISTRIBUTION System operability status OR key parameter indications, plant conditions, and a copy of Tech Specs, DETERMINE if Tech Spec Limiting Condition for Operations have been met, and required actions if any. .1 AC Electrical Distribution .2 Switch Gear Heat Removal

	SRO Q2/77 295004 A2.03

263000.12	263000.12 Given BATTERY & DC DISTRIBUTION System operability status OR key parameter indications, plant conditions, and a copy of Tech Specs, DETERMINE if Tech Spec Limiting Condition for Operations have been met, and required actions if any. .1 AC Electrical Distribution .2 Switch Gear Heat Removal

Other NRC Data

References Provided	Yes - Internal
K/A Justification	This question meets the KA because the examinee has to interpret the battery voltage indications resulting from a partial loss of AC power to the battery charger to determine required actions in Technical Specifications.
SRO-Only Justification	This question is linked to SRO only Task 140109.23 (Apply the administrative requirements for execution of Technical Specifications and Off-Site Dose Calculation Manual Requirements). Also linked to 10CFR55.43(b)(2), Facility operating limitations in the Technical Specifications and their bases.
Additional Information	This question is a high cog question written at the analysis and comprehension level. The examinee has to interpret the indications provided in the stem of the question and then determine required actions based on that analysis (3-SPK).
NRC Exams Only	
Question Type	Bank (CL-ILT-N14077) — **Difficulty** N/A
Technical Reference and Revision #	• ITS 3.8.1 (3.8-2) Amendment No. 141 • ITS 3.8.4 (3.8-24) Amendment No. 187 • ITS 3.8.6 (3.8-30) Amendment No. 142 • OP-CL-108-104-1001 Rev. 11a
Training Objective	DB420001.01.04 Describe the consequences of the following: .3 Loss of AC power on DC electrical loads
Previous NRC Exam Use	ILT 14-1 NRC

2	ID: 2096794	Points: 1.00

The plant was operating at rated thermal power, when the following sequence of events occurred:

Time	Event
1159	A fire occurred in the MCR.
1200	CRS orders the MCR to be evacuated; enters CPS 4003.01 REMOTE SHUTDOWN (RS).
1204	CPS 4003.01 section 4.2.1 Initial MCR Actions prior to Evacuation are completed. MCR Operators begin their transit to the Remote Shutdown Panel (RSP).
1219	MCR Operators reach the RSP to assume control of RPV water level and pressure. The RO reports RPV water level downscale on BOTH WR level instruments and RPV pressure at 580 psig and lowering.

What is the highest Emergency Classification Level that has been exceeded?

A.	HU3
B.	MA5
C.	HA2
D.	HS2

Answer:	D

Answer Justification / Plausibility Statements
D is correct. Per EP-AA-1003 Addendum 3, page CL 2-8, Control of RPV water level has not been reestablished in < 15 minutes, exceeding the HS2 threshold. Per the bases for HS2 (CL 2-132), the time period to establish control of the plant starts when either: • Control of the plant is no longer maintained in the Main Control Room OR • The last Operator has left the Control Room. Since the stem states that the MCR Operators begin their transient at 1204 and reach the RSP at 1219, 15 minutes has elapsed, exceeding the HS2 threshold. Incorrect Responses: A is incorrect but plausible. This answer would be correct if the fire (HU3) was not accompanied by: • damage to a safety system required for the current operating mode (MA5), and • inability to control a key safety function from outside the Control Room (HS2). B is incorrect but plausible. This answer would be correct if the damage to a safety system required for the current operating mode (MA5) was not accompanied by the inability to control a key safety function from outside the Control Room (HS2). C is incorrect but plausible. This answer would be correct if the Control Room evacuation and transfer of plant control to the RSP was not accompanied by the inability to control a key safety function from outside the Control Room (HS2).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295016	B2.1.23	4.3	4.4	1	1	N/A

System Name
Control Room Abandonment

Category Statement
Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q2/77 295016 2.1.23

Other NRC Data

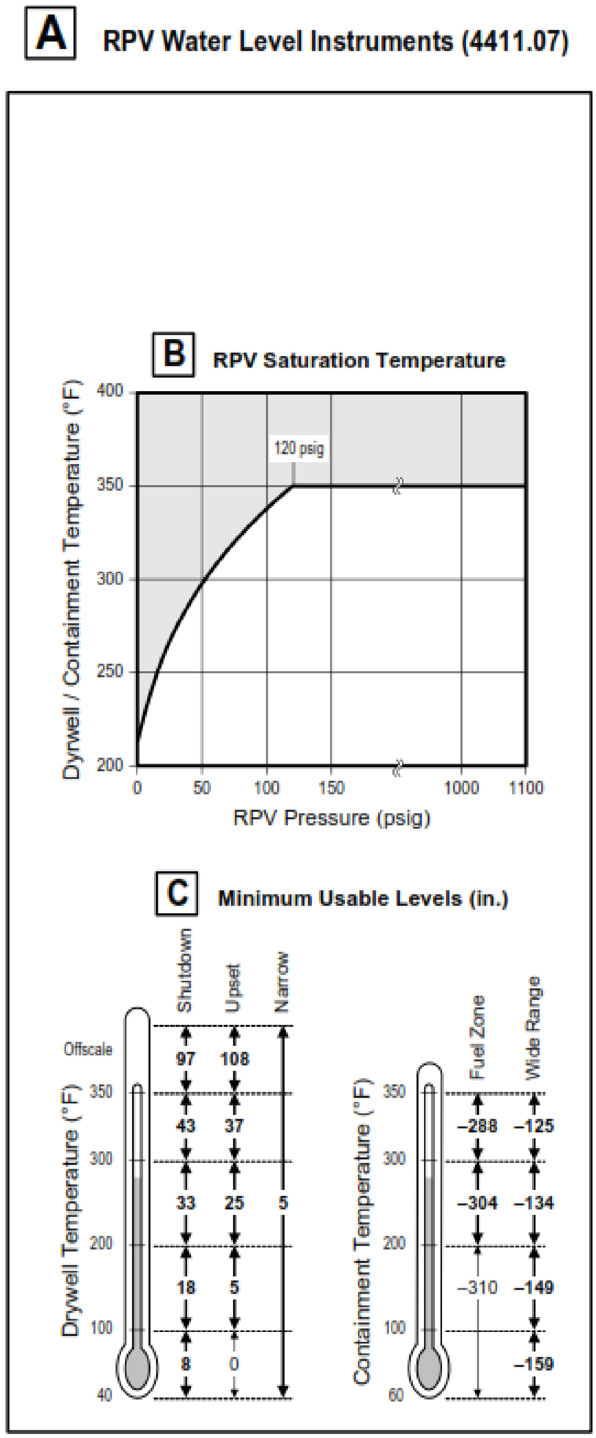
References Provided	EP-AA-1003 Addendum 3 - Pages 2-1 to 2-13 (Hot Matrix) Rev. 2		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to perform integrated plant procedures (E-Plan) during a Control Room Abandonment event to answer the question.		
SRO-Only Justification	This question is linked to 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. It is also linked to SRO only task 997777.01 From the MCR, classify Emergency Action Level IAW EP-AA-1003, Radiological Emergency Plan Annex For Clinton.		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to analyze the conditions in the stem and then determine the appropriate EAL based on the analysis to answer the question (3-SPR/SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-1799571)	**Difficulty**	NA
Technical Reference and Revision #	EP-AA-1003 Addendum 3, Rev. 2		
Training Objective	LP85804.2.4.29 Knowledge of the emergency plan.		
Previous NRC Exam Use	None		

3	ID: 2110191	Points: 1.00

A transient has occurred resulting in a loss of coolant accident requiring a blowdown per EOP-3 Emergency RPV Depressurization, which is complete.

Plant conditions have stabilized.

Given the detail below:


A
RPV Water Level Instruments (4411.07)
B
RPV Saturation Temperature
120 psig
Dywell / Containment Temperature (°F)
400
350
300
250
200
0
50
100
150
1000
1100
RPV Pressure (psig)
C
Minimum Usable Levels (in.)
Shutdown
Upset
Narrow
Offscale
Drywell Temperature (°F)
350
300
200
100
40
97
108
43
37
33
25
5
18
5
8
0
Fuel Zone
Wide Range
Containment Temperature (°F)
350
300
200
100
60
−288
−125
−304
−134
−310
−149
−159

If both Fuel Zone instruments are inoperable, which of the following scenarios require entry into EOP-2 RPV Flooding?

	1B21-R623A Wide Range RPV Level	**1B21-R623B Wide Range RPV Level**	**RPV Pressure**	**DW Temp**	**Cnmt Temp**
Scenario 1	-150" (steady)	Fluctuating -20" to -30"	55 psig	325°F	140°F
Scenario 2	-131" (rising)	-121" (rising)	114 psig	275°F	205°F
Scenario 3	-136" (lowering)	-137" (lowering)	45 psig	305°F	209°F

A.	Scenario 2 ONLY
B.	Scenario 3 ONLY
C.	Scenarios 1 & 3
D.	Scenarios 1 & 2

Answer:	C

Answer Justification / Plausibility Statements
C is correct. Per the level leg overrides in EOP-1 RPV Control and 1A ATWS RPV Control, RPV Flooding is required if RPV water level is unknown.

Per the instructions in Detail A (not provided in the stem) and CPS 4411.07 RPV Level Instrumentation:

- RPV water level instruments **may** be unreliable due to boiling in the instrument legs if drywell or containment temperature is above Fig. B, RPV Saturation Temperature.
- Indications/Contributors of Saturation Failures may include erratic level indication.
- Do not use an RPV water level instrument if level is at or below Fig. C, Minimum Usable Levels.

	> Fig. B?	Saturation Failure?	< Min Usable Level?	Flooding Required?
Scenario 1	DW	Yes	Yes	Yes
Scenario 2	No	No	No (WR)	No
Scenario 3	DW	No	Yes	Yes

Therefore, RPV Flooding is required for scenarios 1 & 3.

Incorrect Responses:

A is incorrect but plausible. This answer would be correct if RPV flooding was required in scenario 2 based solely on the difference in indicated level between 1B21-R623A and B without factoring in trend data (both trending up/NOT fluctuating/erratic).

B is incorrect but plausible. This answer is partially correct in that RPV flooding is required for scenario 3 (WR level below the minimum usable level limit).

D is incorrect but plausible. This answer would be correct if RPV flooding was required in scenario 2 based solely on the difference in indicated level between 1B21-R623A and B without factoring in trend data (both trending up/NOT fluctuating/erratic).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295028.EA2.01	EA2.01	4.0*	4.1*	1		5

System Name
High Drywell Temperature

Category Statement
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
Drywell temperature

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q3/78 295028 A2.01

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to interpret high DW temperature parameters provided in the stem to determine correct procedural actions to answer the question.		
SRO Only Justification Statement:	This question is linked to SRO only task 440301.02 (Enter and Execute EOP-2 Flooding during NON-ATWS conditions) and 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The question is also SRO ONLY level because the evaluation of plant parameters requires knowledge of the usage instructions of details in EOP-1. The interpretation of the data in the table and determining level is unknown to determine an entry condition into EOP-2 is the SRO's job at Clinton. Knowledge of EOP-2 RPV Flooding direct entry conditions is NOT RO level knowledge per the SRO Only clarification guidance.		
Additional Information:	This is a high cog question because the examinee has to analyze conditions provided in the stem, use knowledge and a reference to determine available RPV water level indication and if entry into EOP-2 RPV Flooding is warranted.		
NRC Exams Only (as applicable)			
Question Type:	Modified (CL-ILT-N15091)	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 4411.07 Rev. 6b • CPS 4401.01 Rev. 30		
Training Objective:	LP85804.2.4.9 Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.		
Previous NRC Exam Use:	None		

4	ID: 896372	Points: 1.00

The unit was operating at rated thermal power when severe weather caused a loss of power to the 345KV Switchyard North **AND** South busses and a Main Generator trip.

All appropriate immediate operator actions were taken.

Current plant conditions are as follows:

- SRV 1B21-F051D is full open.
- SRV 1B21-F051C is cycling open and closed.
- Reactor water level is cycling between minus 30" and plus 10”.
- Suppression pool level is 19' 3" and slowly rising.
- Suppression pool temperature is 108°F and slowly rising.

Which of the following actions is required?

The CRS will direct/execute the...

A.	power control actions of CPS 4100.01 Reactor Scram.
B.	level control actions of EOP-1A ATWS RPV Control using Level Band A (Level 3 to 8).
C.	pressure control actions of CPS 4411.09 RPV PRESSURE CONTROL SOURCES section 2.2.2 SRVs.
D.	startup of the Continuous Containment Purge (CCP) system in the Filtered (Auto) Mode IAW CPS 3408.01 Containment Building/Drywell HVAC (VR, VQ).

Answer:	C

Answer Justification / Plausibility Statements

C is correct:

A loss of the 345KV Switchyard North and South Busses will result in a loss of the RAT Transformers A, B, and C and a generator trip. This will result in a loss of BOP power in the plant, removing the Main Condenser as a heat sink.

The current conditions provided in the stem indicate that an ATWS has occurred. Each SRV can pass approximately 6% reactor power, so with one SRV full open and the second cycling open and closed, reactor power is between 6 and 12%. Per the pressure leg of EOP-1A ATWS RPV Control, stabilize RPV pressure below 1065 psig using main turbine bypass valves. Use other methods below if needed. Since the Main Condenser is not available due to the loss of BOP power, SRVs are required to be used to control reactor pressure. This is accomplished by executing the pressure control actions of CPS 4411.09 RPV Pressure Control Sources section 2.2.2 SRVs.

Incorrect Responses:

A is incorrect but plausible. With reactor power above 5%, the power leg of EOP-1A directs inserting control rods per CPS 4411.08 Alternate Control Rod Insertion, not CPS 4100.01 Reactor Scram. This answer would be correct if reactor power was less than IRM Range 7 and lowering and no boron had been injected.

B is incorrect but plausible because EOP-1A directs controlling RPV water level using Level Band A during low power (< 5%) ATWS conditions. With power in the question stem > 5%, Level Band B is used.

D is incorrect but plausible because CPS 4100.01 Reactor Scram step 4.4.2 directs shifting CCP to Filtered Mode if any SRVs have lifted or are cycling. Incorrect because the VR/VQ fans used for CCP will be deenergized due to the loss of non-vital power in the stem. Impacted fans and power supplies are as follows:

VR/VQ Fans (CPS 3408.01 section 8.1.1.2)	VR/VQ Fan Power Sources (CPS 3408.01E001	Unit Substation Power Sources (CPS 3502.01 Att. 1)
1VR07CA	480V Unit Substation K (0AP52E)	4160V Bus 1A
1VR07CB	480V Unit Substation L (0AP53E)	4160V Bus 1B
1VR06CA	480V Unit Substation K (0AP52E)	4160V Bus 1A
1VR06CB	480V Unit Substation L (0AP53E)	4160V Bus 1B
0VQ02CA	480V Unit Substation 1F (1AP16E)	6900V Bus 1A
0VQ02CB	480V Unit Substation 1G (1AP17E)	6900V Bus 1B

4160V Bus 1A/1B and 6900V Bus 1A/1B are non-vital buses power from the Unit Auxiliary Transformers when the Main Generator is online, and from the RAT transformers when the Main Generator is offline.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295025	B2.4.06	3.7	4.7	1	1	N/A

System Name
High Reactor Pressure

Category Statement
Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q4/79 295025 2.4.6

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of EOP mitigating strategies (e.g. RPV pressure control strategies) during an ATWS to answer the question.
SRO-Only Justification	This question is linked to SRO only task 440401.02 (Enter and Execute EOP-1A when required) and 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Additional Information	This is a high cog question written and the analysis and application level. The examinee has to analyze conditions in the stem and then determine required actions based on that analysis (3-SPK).

NRC Exams Only

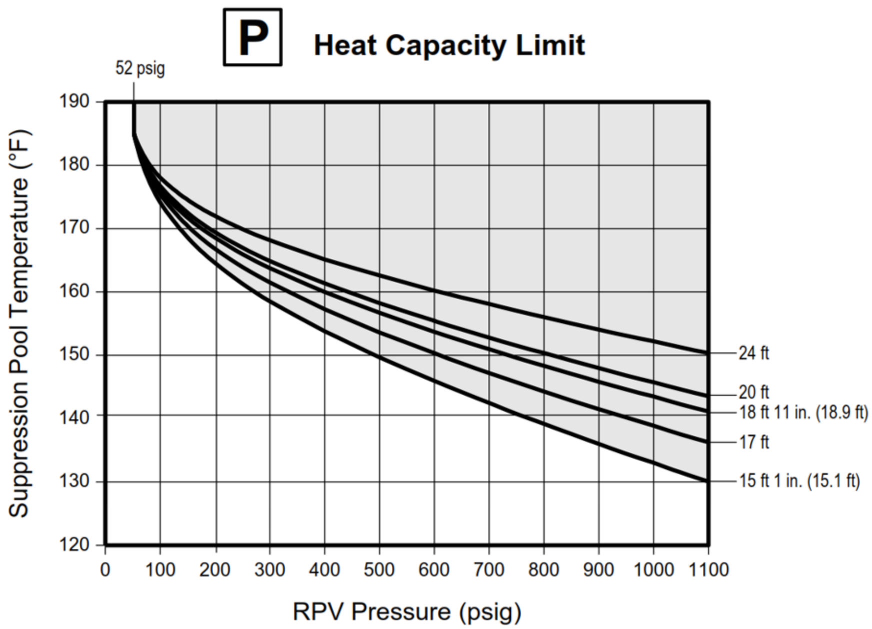
Question Type	Bank (CL-ILT-N12081)	**Difficulty**	NA

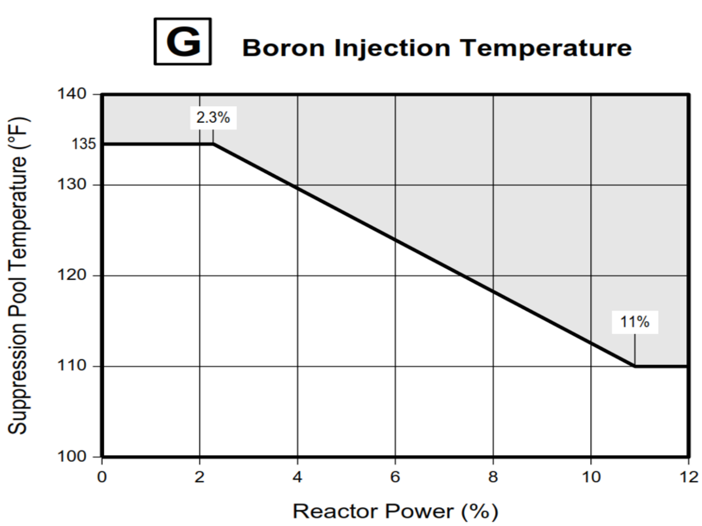
Technical Reference and Revision #	• CPS 4404.01 ATWS RPV Control Rev. 30 • CPS 4411.09 Rev. 6a • CPS 4100.01 Rev. 23f • CPS 3408.01 Rev. 20e • CPS 3408.01E001 Rev. 11a • CPS 3502.01 Rev. 9b
Training Objective	N-CL-OPS-DB-LP87553.01.03 Given a diagram of EOP-1A: .04 List the available methods to be used to stabilize/control RPV pressure in EOP-1A
Previous NRC Exam Use	CPS ILT 12-1 NRC exam

5	ID: 2096815	Points: 1.00

An ATWS is in progress, with the following conditions:

- Reactor power is 35%.
- SRVs are cycling.
- Reactor pressure is about 950 psig.
- Reactor water level is +10 inches.
- Suppression Pool Level is 19 feet.
- Suppression Pool Temperature is 143°F and slowly rising.





The next required action is to...

A.	lower level to reduce subcooling (Level Band B).
B.	lower pressure to stay below the Heat Capacity Limit.
C.	enter EOP-3 Emergency Depressurization and blowdown.
D.	rapidly depressurize the RPV (greater than 100°F per hour) using main turbine bypass valves.

Answer:	B

Answer Justification / Plausibility Statements
B is correct: The pool temperature leg of EOP-6 Primary Containment Control requires RPV pressure to be lowered to stay below the limit of Fig. P Heat Capacity Limit. Incorrect Responses: A is incorrect but plausible. During an ATWS, RPV water level is controlled by 1 of 3 level legs (Level Band A, B, or C). With the conditions of the If-AND-AND-AND-THEN override met, Level Band C is used to reduce reactor power. Level Band B (lower level to reduce subcooling) is incorrect because the conditions of the If-AND-AND-AND-THEN override are met C is incorrect but plausible because heat is being added to the Suppression Pool and threatening to exceed the Heat Capacity Limit. The next required action in EOP-6 Primary Containment Control Suppression Pool Temperature leg is to lower RPV pressure to stay below HCTL. Performing a blowdown is only required if attempts at reducing reactor pressure failed to control Suppression Pool temperature and RPV pressure below HCTL. D is incorrect but plausible. Anticipating blowdown is required by EOP-6 Primary Containment Control if SP temperature is above the HCTL, but only when in EOP-1 RPV Control. Performing anticipatory blowdowns is not appropriate in EOP-1A ATWS RPV Control when the pressure leg directs stabilizing reactor pressure.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295026.EA2.03	EA2.03	3.9	4.0	1		5

System Name
Suppression Pool High Water Temperature

Category Statement
Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
Reactor pressure

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q5/80 295026 A2.03

LP87553.01.03	N-CL-OPS-DB-LP87553.01.03 Given a diagram of EOP-1A: .01 Describe the conditions to exit/transfer from EOP-1A .02 State the bases for each individual step/action of EOP-1A .03 Discuss the importance of verifying that the appropriate automatic actions of EOP-1A occur as required .04 List the available methods to be used to stabilize/control RPV pressure in EOP-1A .05 List the Preferred ATWS level control systems.

Other NRC Data

References Provided	Yes - Internal		
K/A Justification	This question meets the KA because the examinee has to interpret the High Suppression Pool Temperature and reactor pressure conditions provided in the stem and then determine the actions required to mitigate the condition to answer the question.		
SRO-Only Justification	SRO Justification - this question is linked to 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		
Additional Information	Cog Level Justification - this question is high cog because the examinee has to solve a problem based on knowledge of the conditions provided in the stem (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A14078)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4402.01 Rev. 30 • CPS 4404.01 Rev. 30		
Training Objective	N-CL-OPS-DB-LP87558.01.17 Given a condition resulting in approaching the Heat Capacity Limit, Fig. P, and a diagram of EOP-6, determine when it would be appropriate to Blowdown.		
Previous NRC Exam Use	None		

6	ID: 2106250	Points: 1.00

The plant was operating at rated thermal power when an event occurred causing Suppression Pool level to lower.

Suppression Pool level is currently 13.1 feet and lowering.

Assuming all required EOP actions have been taken up to this point, what is the bases for the NEXT required action as suppression pool level continues to lower?

The next required action will...

A.	prevent direct discharge of SRVs into the containment airspace.

B.	protect the ECCS pumps from inadequate net positive suction head (NPSH).

C.	prevent direct pressurization of the containment through the Hydrogen Mixing Compressor discharge lines.

D.	add a significant volume of water to the Suppression Pool to delay the requirement to perform an emergency depressurization.

Answer:	C

Answer Justification / Plausibility Statements
C is correct: EOP-6 Primary Containment Control requires the Hydrogen Mixing Compressors to be stopped when Suppression Pool level drops below 13.1 ft. Per the EOP-TB (page 8-21), this action is required because the mixer discharge would be uncovered and mixer operation would open a path from the drywell directly into the containment. A break inside the drywell could then overpressurize the containment. The direction to stop mixers in the Suppression Pool Level branch ensures that appropriate action is taken if suppression pool level drops below the discharge elevation after the mixers are started. Incorrect Responses: A is correct but plausible because EOP-3 prohibits initiating ADS (opening SRVs) with the RPV pressurized and SP level at or below 8 feet, but incorrect because ADS is initiated with Suppression Pool level at or above 15.1 feet. B is incorrect but plausible. The NPSH of the ECCS Pumps is impacted by lowering suppression pool level. Detail Z NPSH / Vortex Limits states that the low suppression pool level limit for HPCS, LPCS, and RHR is 11 ft. Incorrect because Detail Z provides a caution that damage could occur, but does not prohibit pump operation. D is incorrect but plausible because dumping the upper pools will result in a 2' increase in SP level, which will delay the requirement to perform a blowdown and allow additional time to attempt to arrest the lowering suppression pool level condition. Incorrect because this action is performed before suppression pool level drops to 15.1 feet. Based on the conditions in the stem, this action would already have been performed.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295030	B2.4.18	3.3	4.0	1	1	N/A

System Name
Low Suppression Pool Water Level

Category Statement
Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.1	43.1

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q6/81 295030 2.4.18

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because it requires knowledge of the EOP bases for lowering Suppression Pool level to answer the question.		
SRO-Only Justification	This question is linked to SRO only task 440201.12 Respond to a Low Suppression Pool Level per EOP-6 and 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.		
Additional Information	Cog Level Justification - this is a high cog question written at the analysis/application level. The examinee has to analyze plant conditions presented in the stem, and then determine the bases for the correct actions to answer the question (3-SPK)		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• EOP-TB Rev. 07 • CPS 4402.01 Rev. 30 • CPS 4407.01 Rev. 30		
Training Objective	N-CL-OPS-DB-LP87558.01.02 Given a diagram of EOP-6: .02 State the bases for each individual step/action of EOP-6		
Previous NRC Exam Use	None		

7	ID: 2096835	Points: 1.00

The plant was operating at rated thermal power when an inadvertent Group 1 Isolation and ATWS occurred.

Appropriate immediate actions were completed.

Current plant conditions:

Parameter	Value	Trend
Reactor Power	35%	stable
RPV Water Level	30 inches	stable
Suppression Pool Temperature	111°F	rising
DW pressure	1.9 psig	rising

The MCR crew has terminated and prevented injection from detail F1 systems.

The earliest time you will direct RPV injection to recommence is _____.

Time	SRV Status	Reactor Power (%)	RPV Water Level (inches)	DW Pressure (psig)
0100	open	35	30	1.9
0105	open	30	25	1.7
0110	open	25	0	1.6
0111	open	10	-60	1.6
0112	open	4	-100	1.6
0113	shut	1	-140	1.5
0115	shut	IRM Rg. 8	-160	1.4

A.	0110

B.	0111

C.	0112

D.	0113

Answer:	C

Answer Justification / Plausibility Statements
C is correct: Per EOP-1A ATWS RPV Control, the conditions of the IF-AND-AND-AND-THEN override in the RPV Water Level Leg have been satisfied, requiring RPV water level to be lowered to reduce Rx power. After injection has been terminated and prevented, RPV water level is lowered until: • Reactor power is below 5%, OR • RPV water level drops to –140 in., OR • All SRVs stay closed and drywell pressure stays below 1.68 psig Reactor power reaches 4% at 0112. Incorrect Responses: A is incorrect but plausible. At 0110, DW pressure meets the re-injection criteria, but SRVs are still open - re-injection is not allowed. B is incorrect but plausible. At 0111, RPV level drops to -60 inches, which meets the re-injection criteria for Level Band B (level lowered to reduce subcooling). For level band C, the RPV level criterial is either -140" or the level at which Reactor power is below 5%. This first occurs at 0112. D is incorrect but plausible. At 0113, SRVs are shut with DW pressure below 1.68 psig, which meets the re-injection criteria, but this occurs after Reactor power reaches 4%.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295031.EA2.02	EA2.02	4.0	4.2*	1		2

System Name
Reactor Low Water Level

Category Statement
Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
Reactor power

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

LP87553.01.12	N-CL-OPS-DB-LP87553.01.12 Given a diagram of EOP-1A, appropriate EOP-support procedures, and simulated plant conditions determine which course of action should take priority over others: .01 Establishing RPV water level in accordance with the Level Bands provided in the Level Path. .02 Preventing LPCS/LPCI injection not needed for core cooling if drywell pressure is above 1.68 psig .03 Defeating interlocks for inserting control rods using CPS No. 4411.08, ALTERNATE CONTROL ROD INSERTION .04 Use of CPS No. 4100.01, REACTOR SCRAM vs. CPS No. 4411.08, ALTERNATE CONTROL ROD INSERTION, to insert control rods

	Q7/82 295031 A2.02

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to interpret reactor power when establishing re-injection during an ATWS with lowering RPV level to answer the question.		
SRO-Only Justification	This question is linked to SRO only task 440401.04 (440401.04) Determine when to initiate RPV injection with ATWS injection systems for EOP-1A and to 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to analyze the conditions in the stem and then determine required actions based on the analysis (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-LC-0978)	**Difficulty**	NA
Technical Reference and Revision #	CPS 4404.01 Rev. 30		
Training Objective	N-CL-OPS-DB-LP87553.01.12 Given a diagram of EOP-1A, appropriate EOP-support procedures, and simulated plant conditions determine which course of action should take priority over others: .01 Establishing RPV water level in accordance with the Level Bands provided in the Level Path.		
Previous NRC Exam Use	None		

8	ID: 2096836	Points: 1.00

The Unit was operating at rated thermal power when Main Condenser vacuum began to degrade.

- Air in leakage to the Main Condenser is approximately 80 scfm and is slowly rising.
- Main Condenser vacuum is currently at 26.0” Hg and continues to degrade.

Which of the following actions should the CRS next direct, in accordance with CPS 4004.02 Loss of Vacuum?

A.	Initiate a Rapid Plant Shutdown.
B.	Place both SJAE Trains in service.
C.	Start an additional Circulating Water Pump.
D.	Perform an Emergency Power Reduction to 80%.

Answer:	D

Answer Justification / Plausibility Statements
D is correct: CPS 4004.02 Loss of Vacuum, Section 4.1 Low / Deteriorating Vacuum, subsequent action step 4.1.1 requires rapidly lowering reactor power to ~ 80% per CPS 3005.01, Unit Power Changes Section 8.2.3 (Emergency Power Reduction). Incorrect Responses: A is incorrect but plausible. CPS 4004.02 step 4.1.2 requires a Rapid Plant Shutdown to be performed if vacuum cannot be controlled above 24" after performing the Emergency Power Reduction to 80%. B is incorrect but plausible. Placing a second SJAE Train in service will theoretically remove more air from the Main Condenser but is not procedurally permitted. C is incorrect but plausible. This answer would be correct if the loss/deteriorating vacuum was caused by inadequate Circulating Water flow through the Main Condenser. CPS 4004.02 step 4.1.6.2 directs referring to CPS 3113.01 Circulating Water (CW) under low/deteriorating vacuum conditions. Since the loss of vacuum is being caused by high air inleakage, starting an additional CW Pump will not correct the low vacuum condition.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295002.AA2.01	AA2.01	2.9	3.1	1		3

System Name
Loss of Main Condenser Vacuum

Category Statement
Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM : (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
Condenser vacuum/absolute pressure

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

DB400402-A.01.01	DB400402-A.01.01 Given specific plant conditions, determine if CPS No. 4004.02, LOSS OF CONDENSER VACUUM should be used.

CPS.315470	(400402.01) respond to a loss of main condenser vacuum.

	SRO Q8/83 295002 A2.01

	Q8/83 295002 A2.01

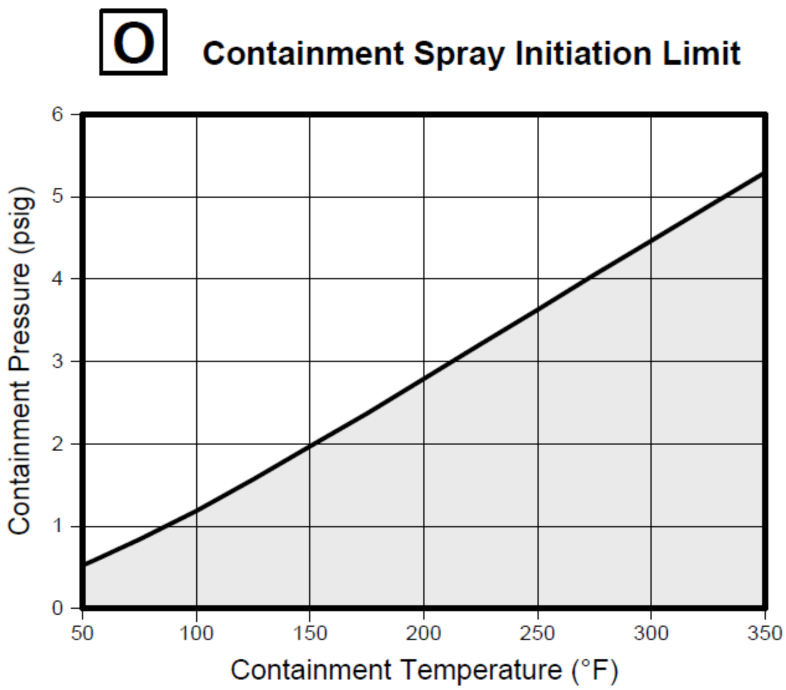
Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to interpret the deteriorating vacuum conditions in the stem and determine corrective actions to answer the question.		
SRO-Only Justification	This question is linked to 10CFR55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Specifically, the question involves the assessment of plant conditions as presented and then selecting the section of a procedure (CPS 4004.02 Section 4.1) with which to proceed.		
Additional Information	Cog Level Justification - this is a low cog question written at the memory level. The examinee has to recall actions contained in a procedure for low/deteriorating Main Condenser Vacuum.		
NRC Exams Only			
Question Type	Bank (CL-ILT-N15083)	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 4004.02 Rev. 7 • CPS 3005.01 Rev. 43f		
Training Objective	DB400402-A.01.03 Describe the operational implications on condenser vacuum of the following : .3 Condenser Vacuum can not be controlled above 25” HG VAC.		
Previous NRC Exam Use	ILT 15-1 NRC Exam		

9	ID: 2105805	Points: 1.00

A LOCA has occurred causing the trends shown below.

Reactor Pressure (psig)	WR RPV Water Level (inches)	Containment Pressure (psig)	Containment Temperature (°F)	DW Pressure (psig)	DW Temp (°F)
767 (trending down)	-112 (trending down)	2.2 (trending up)	146 (trending up)	4.5 (trending up)	163 (trending up)



Appropriate EOPs have been entered and immediate actions performed.

Which of the following actions is required?

A.	Initiate Containment Sprays.
B.	Prevent LPCS and LPCI injection.
C.	Perform CPS 4410.00C006 Defeating VP/WO Interlocks.
D.	Enter EOP-3 Emergency RPV Depressurization and perform an emergency depressurization.

Answer:	C

Answer Justification / Plausibility Statements
C is correct - performance of CPS 4410.00C006 is required by EOP-6 Primary Containment Control if DW temperature cannot be maintained below 150°F. Incorrect Responses A is incorrect but plausible. Initiation of containment sprays is performed only if in the "OK to Spray" region of Figure O Containment Spray Initiation Limit <u>and</u> RHR Pumps are not needed for core cooling. With RPV level at -112 inches, RHR pumps are needed for core cooling per the guidelines of OP-CL-101-111-1001 Strategies for Successful Transient Mitigation. B is incorrect but plausible. This action is performed before RPV pressure reaches 472 psig but not if RHR Pumps are needed for core cooling. With RPV level at -112 inches, RHR pumps are needed for core cooling. Per the guidelines of OP-CL-101-111-1001 Strategies for Successful Transient Mitigation, RPV injection sources (i.e., Residual Heat Removal (RHR), Low Pressure Core Spray (LPCS), etc.) are required to be maximized (aligned and capable of injection) when RPV level is below TAF, **regardless of RPV pressure**. D is incorrect but plausible. Blowdown is required if DW temperature cannot be lowered and maintained below 330°F or containment temperature cannot be held below 185°F.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295009	B2.4.20	3.8	4.3	1	2	N/A

System Name
Low Reactor Water Level

Category Statement
Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q9/84 295009 2.4.20

Other NRC Data

References Provided	Yes - Internal		
K/A Justification	This question meets the KA because the examinee has to analyze the parameters in the stem, and then determine required actions based on a note in EOP-6 Primary Containment Control Drywell Temperature leg that drywell temperature affects RPV water level indication - Check Detail A.		
SRO-Only Justification	This question is linked to SRO only task 100509.05 Determine when an EOP Support Procedure is to be performed. Also linked to 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		
Additional Information	Cog Level Justification - this is a high cog question written at the analysis and application level because the examinee has to analyze the data provided in the question stem, and then determine required actions based on that analysis (3-SPK).		
NRC Exams Only			
Question Type	New	**Difficulty**	NA
Technical Reference and Revision #	• OP-CL-101-111-1001 Rev. 15b • CPS 4402.01 Rev. 30 • CPS 4401.01 Rev. 30		
Training Objective	N-CL-OPS-DB-LP87558.01.03 Given an EOP-6 condition, discuss the consequences/impact of the following: .03 Rising Containment and/or Drywell temperature on RPV water level instrumentation		
Previous NRC Exam Use	None		

10	ID: 2096849	Points: 1.00

The plant was operating at rated thermal power when the following sequence of events occurred:

Time	Event
0115	Annunciator 5040-2F Low Level Spent Fuel Stor Pool received.
0120	Annunciator 5040-3F Low-Low Level Spent Fuel Stor Pool received.
0130	High alarm is received on 1RIX-AR016 Spent Fuel Storage Area.
0140	A Group 19 Secondary Containment Isolation occurs due to Fuel Building Exhaust High Radiation. The Standby Gas Treatment System initiates.
0150	RP reports that radiation levels in the hallways: • outside the Fuel Pool Cooling (FC) Pump Room is 30 R/hr and rising, AND • outside the Fuel Pool Cooling (FC) Heat Exchanger rooms is 30 R/hr and rising.

Repair teams are attempting to isolate the leak.

Operators are aligning systems to refill the Spent Fuel Storage Pool.

U Area Radiation Limits

Area	Method	Max Normal	Max Safe
HPCS Pump Room	Survey	10 mr/hr	25 R/hr
Aux Bldg Aisle El 707' 6" (West)	Survey	10 mr/hr	25 R/hr
RHR Pump Room A RHR Heat Exch Room A (under HX)	Survey Survey	100 mr/hr 100 mr/hr	25 R/hr 25 R/hr
RHR Pump Room B RHR Heat Exch Room B (under HX)	Survey Survey	100 mr/hr 100 mr/hr	25 R/hr 25 R/hr
RHR Pump Room C	Survey	150 mr/hr	25 R/hr
RCIC Pump Room RCIC Instr Pnl Rm	Survey Survey	20 mr/hr 20 mr/hr	400 R/hr 400 R/hr
LPCS Pump Room	Survey	30 mr/hr	25 R/hr
Aux Bldg Access Aisle EL 737' (West)	Survey	10 mr/hr	25 R/hr
Aux Bldg Access Aisle EL 737' (East)	Survey	10 mr/hr	25 R/hr
MSIV-LCS Blower Rooms Aux Bldg Below MS Tunnel	Survey Survey	30 mr/hr 60 mr/hr	400 R/hr 400 R/hr
RWCU Pump Room A (inner door)	Survey	100 mr/hr	400 R/hr
RWCU Pump Room B (inner door)	Survey	100 mr/hr	400 R/hr
RWCU Pump Room C (inner door)	Survey	100 mr/hr	400 R/hr
Aux Bldg Steam Tunnel (entrance)	Survey	100 mr/hr	400 R/hr
Fuel Pool Clg Heat Exch Rm	Survey	100 mr/hr	400 R/hr
Fuel Bldg Gen Area El 712'	Survey	2.5 mr/hr	25 R/hr
Fuel Bldg Pipe Valve Room	Survey	10 mr/hr	400 R/hr
Fuel Bldg Fuel Pool Clg Pmp Rm	Survey	20 mr/hr	400 R/hr
Fuel Bldg Gen Area El 737'	Survey	2.5 mr/hr	25 R/hr
Aux Bldg Gas Cont Boundary (El 762' West)	Survey	10 mr/hr	25 R/hr
Aux Bldg Gas Cont Boundary (El 762' East)	Survey	10 mr/hr	25 R/hr
Fuel Bldg Gen Area 755' Spent Fuel Storage Area New Fuel Storage Vault Fuel Bldg Fuel Handling Platform	Survey 1RIX-AR016 (5140.08) 1RIX-AR019 (5140.10) 1RIX-AR052 (5140.13) 1RIS-AR024	2.5 mr/hr 2.5 mr/hr 2.5 mr/hr 2.5 mr/hr 10 mr/hr	25 R/hr N/A N/A N/A N/A

Which of the following actions must be performed NEXT?

A.	Enter CPS 3006.01 Unit Shutdown and perform a normal plant shutdown.

B.	Scram the reactor, enter EOP-1 RPV Control and then lower and maintain reactor pressure at 500 - 600 psig.

C.	Scram the reactor, enter EOP-1 RPV Control and then rapidly depressurize the reactor using the Main Turbine Bypass Valves.

D.	Scram the reactor, enter EOP-1 RPV Control and EOP-3 Emergency RPV Depressurization and perform a blowdown.

Answer:	A

Answer Justification / Plausibility Statements
A is correct. Per EOP-8 Secondary Containment Control, actions are required in the following two legs: • Temperature/Radiation - with a non-primary system discharging into the secondary containment and with 2 or more areas with radiation levels having exceeded max safe values (25 R/hr in the Fuel Bldg Gen Area El 712' and Fuel Bldg Gen Area El 737'), EOP-8 requires a reactor shutdown IAW CPS 3006.01 Unit Shutdown. • Spent Fuel Pool Level - contains actions to restore Spent Fuel Pool Level. Those actions are in progress according to the stem. Incorrect Responses: B is incorrect but plausible. This action is required if a primary system is discharging into the secondary containment and the discharge cannot be isolated. The spent fuel pool does not communicate with the RPV and therefore is not a primary system. Lowering reactor pressure to 500-600 psig is directed post scram by OP-CL-101-111-1001 Strategies for Successful Transient Mitigation to minimize the driving head behind a primary leak. C is incorrect but plausible. This action is required if a primary system is discharging into the secondary containment, the discharge cannot be isolated, and 2 or more areas are approaching max safe values of the same parameter. Anticipating blowdown is not appropriate in this scenario because lowering RPV pressure will not impact the leak rate from the Spent Fuel Pool. D is incorrect but plausible. This action is required if a primary system is discharging into the secondary containment, the discharge cannot be isolated, and 2 or more areas have exceeded max safe values of the same parameter. Performing a blowdown is not appropriate in this scenario because lowering RPV pressure will not impact the leak rate from the Spent Fuel Pool.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295017.AA2.03	AA2.03	3.1	3.9	1		9

System Name
High Off-Site Release Rate

Category Statement
Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : (CFR: 41.10 / 43.5 / 45.13)

K/A Statement
†Radiation levels: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

LP87559.01.05	N-CL-OPS-DB-LP87559.01.05 Explain when a decision should be made to Scram and enter EOP-1 due to the following: .01 Temperature, radiation level or water level approaching a maximum safe value in any one area. .02 A primary system is discharging into Secondary Containment and cannot be isolated.

LP87559.01.06	N-CL-OPS-DB-LP87559.01.06 Explain when a decision should be made to Blowdown due to the following: .01 Temperature, radiation level or water level approaching a maximum safe value in two or more areas. .02 A primary system is discharging into Secondary Containment and cannot be isolated.

	SRO Q9/84 295017 2.4.6

	Q10/85 295017 A2.03

Other NRC Data

References Provided	Yes - Internal
K/A Justification	This question meets the KA because the examinee has to interpret the radiation levels provided in the stem and then determine the EOP-8 actions required during a high off-site release rate event.
SRO-Only Justification	This question is linked to linked to 10CFR55.73(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowing the EOP entry conditions does not help in answering this question. Directing EOP actions at the end of an EOP flowchart is an SRO only function.
Additional Information	Cog Level Justification - this is a high cog question written at the analysis & comprehension level. The examinee has to analyze the data provided in the stem and then determine required actions based on that analysis and knowledge of EOP required actions.

NRC Exams Only

Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	CPS 4406.01 Rev. 30		
Training Objective	N-CL-OPS-DB-LP87559.01.05 Explain when a decision should be made to Scram and enter EOP-1 due to the following: .01 Temperature, radiation level or water level approaching a maximum safe value in any one area. .02 A primary system is discharging into Secondary Containment and cannot be isolated.		
Previous NRC Exam Use	None		

11	ID: 2106654	Points: 1.00

With the Reactor System Mode Switch in the Start & Hot Stby position, a rod block will occur if _____(1)_____ SRMs experience an upscale failure.

Technical Specification B3.3.1.2 Source Range Monitor (SRM) Instrumentation bases will permit a reactor startup to continue for 4 hours with a minimum of _____(2)_____ SRMs operable.

A.	(1) 1 of 4 (2) 1 of 4

B.	(1) 1 of 4 (2) 3 of 4

C.	(1) 2 of 4 (2) 1 of 4

D.	(1) 2 of 4 (2) 3 of 4

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Part 1 - Per CPS 3304.02 Table 1: Rod Block Troubleshooting Guide, any SRM upscale or inoperable signal will result in a rod block and receipt of annunciator 5006-2H Rod Out Block. Part 2 - Per ITS B3.3.1.2 SRM Instrumentation (page B3.3-33) discussion of Actions A.1 and B.1, providing that at least one SRM remains OPERABLE, Required Action A.1 allows 4 hours to restore the required SRMs to OPERABLE status. During this time, control rod withdrawal and power increase are not precluded by this Required Action. Having the ability to monitor the core with at least one SRM, proceeding to IRM Range 3 or greater and thereby exiting the Applicability of this LCO, is acceptable for ensuring adequate core monitoring and allowing continue operation. Incorrect Responses: B is incorrect but plausible. Part 1 is correct. Part 2 is plausible because ITS 3.3.1.2 requires suspension of control rod withdrawals if 3 required SRMs are inoperable in Mode 2 with IRMs on Range 2 or below. Incorrect because 1 SRM channel is required to be operable; 3.3.1.2 Condition B applies when there are zero OPERABLE SRMs. C is incorrect but plausible and would be correct if SRM instrumentation was treated similarly to RPS instrumentation in Technical Specifications. ITS 3.3.1.1 RPS Instrumentation recognizes the 2 out of 4 logic configuration for RPS and allows operation for up to 6 hours with two RPS channels inoperable. Part 2 is correct. D is incorrect but plausible: • and would be correct if SRM instrumentation was treated similarly to RPS instrumentation in Technical Specifications. ITS 3.3.1.1 RPS Instrumentation recognizes the 2 out of 4 logic configuration for RPS and allows operation for up to 6 hours with two RPS channels inoperable. • Part 2 is plausible because ITS 3.3.1.2 requires suspension of control rod withdrawals if 3 required SRMs are inoperable in Mode 2 with IRMs on Range 2 or below. Incorrect because 1 SRM channel is required to be operable; 3.3.1.2 Condition B applies when there are zero OPERABLE SRMs.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
215004.A2.02	A2.02	3.4	3.7	2		7

System Name
Source Range Monitor (SRM) System

Category Statement
Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

K/A Statement
SRM inop condition

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q11/86 215004 A2.02

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to predict the impacts of an SRM inop condition, and then use procedures to mitigate the consequences to answer the question.
SRO-Only Justification	This question is linked to linked to 10CFR55.73(b)(2) - Facility operating limitations in the technical specifications and their bases.
Additional Information	This is a low cog question written at the memory level. The examinee has to recall facts in ITS and annunciator procedures to answer the question (1-F/1-P).

NRC Exams Only			
Question Type	New	**Difficulty**	NA

Technical Reference and Revision #	• CPS 3304.02 Rev. 22e • CPS 5006.02 (2H) Rev. 28d • ITS 3.3.1.2 (3.3-10 and 3.3-14) Amendment No. 95 • ITS B3.3.1.2 (B3.3-33) Revision No. 0 • ITS B3.3.1.2 (B3.3-34) Revision No. 1-1
Training Objective	215004.06 Given an Source Range Monitor System Annunciator, DESCRIBE: a. The condition causing the annunciator b. Any automatic actions c. Any operational implications
Previous NRC Exam Use	None

12	ID: 2106287	Points: 1.00

The plant is operating at 18% power.

Numerous LPRMs failed downscale (marked with an asterisk and shaded) due to a reactor coolant chemistry excursion.

Div 1 APRM Channel A LPRM Detectors			
06-23A *	06-15B	14-15C	14-07D
06-39A *	06-31B	14-31C *	14-23D
22-07A *	22-15B *	14-47C	14-39D
22-23A *	22-31B	30-15C	30-07D
22-39A *	22-47B	30-31C	30-23D
38-07A *	38-15B *	30-47C	30-39D *
38-23A *	38-31B	46-15C *	46-23D
38-39A *	38-47B	46-31C	46-39D *
		47-47C *	

Div 2 APRM Channel B LPRM Detectors			
14-07A *	06-23B *	06-15C	14-15D *
14-23A *	06-39B	06-31C *	14-31D
14-39A	22-07B	22-15C *	14-47D *
30-07A	22-23B *	22-31C	30-15D
30-23A *	22-39B	22-47C	30-31D *
30-39A	38-07B *	38-15C *	30-47D
46-23A *	38-23B *	38-31C	46-15D
46-39A	38-39B	38-47C *	46-31D *
			46-47D

Div 3 APRM Channel C LPRM Detectors			
14-15A *	14-07B *	06-23C	06-15D *
14-31A *	14-23B	06-39C *	06-31D
14-47A	14-39B	22-07C	22-15D *
30-15A	30-07B	22-23C *	22-31D *
30-31A	30-23B *	22-39C	22-47D
30-47A *	30-39B *	38-07C *	38-15D
46-15A	46-23B	38-23C	38-31D *
46-31A *	46-39B *	38-39C *	38-47D *
46-47A *			

Div 4 APRM Channel D LPRM Detectors			
06-15A *	14-15B *	14-07C *	06-23D
06-31A *	14-31B *	14-23C	06-39D
22-15A	14-47B	14-39C *	22-07D
22-31A *	30-15B	30-07C	22-23D *
22-47A *	30-31B	30-23C	22-39D
38-15A	30-47B *	30-39C	38-07D
38-31A *	46-15B	46-23C	38-23D
38-47A *	46-31B *	46-39C	38-39D
	46-47B		

NO LPRMs are bypassed.

Based solely on LPRM status, required action(s) is/are to declare _____ of the 4 APRM channels inoperable.

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

Answer:	B

Answer Justification / Plausibility Statements
B is correct. Per ITS B3.3.1.1 RPS Instrumentation, the APRM System is composed of four channels, each providing an input to each of the four RPS trip logic divisions. All four Average Power Range Monitor Neutron Flux-High, Setdown channels are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 16 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located. Based on the above information, the SRO will make the following APRM operability call: • APRM 'A' is inoperable due to having less than two operable LPRM inputs on axial level 'A'. • APRM 'B' is OPERABLE (has at least 2 LPRM inputs per level and 17 operable LPRM inputs total). • APRM 'C' is inoperable due to having less than 16 operable LPRM inputs. • APRM 'D' is OPERABLE (has at least 2 LPRM inputs per level and 20 operable LPRM inputs total). Incorrect Responses: A is incorrect but plausible. This answer would be correct if ITS B3.3.1.1 RPS Instrumentation did not have a required minimum number of LPRM inputs from each of the four axial levels. C is incorrect but plausible because APRM operability is determined in part by the status of the associated LPRMs. With the larger number of inop LPRMs in Divisions 1, 2, and 3 it is plausible that 3 channels could be considered inoperable. D is incorrect but plausible because APRM operability is determined in part by the status of the associated LPRMs. With the large number of inop LPRMs in Divisions 1, 2, 3, and 4, it is plausible that 4 channels could be considered inoperable.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.215005	B2.2.25	3.2	4.2	2	1	N/A

System Name
Average Power Range Monitor/Local Power Range Monitor System

Category Statement
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5
41.7	41.7

10CFR55-43b (SRO) Data

Para Num	Text
43.2	43.2

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q12/87 215005 2.2.25

215005.14	215005.14 Given AVERAGE POWER RANGE MONITOR System operability status and a copy of Tech Specs, DISCUSS the bases for the AVERAGE POWER RANGE MONITOR System Tech Spec LCO, related safety limits and Limiting Safety System Settings

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee must have knowledge of the bases for TS LCO 3.3.1.1 RPS Instrumentation in order to answer this question.		
SRO-Only Justification	This question presents a plant condition that the SRO must recognize and apply Tech Spec requirements to determine the required action to place the plant into an acceptable condition. Linked to 10CFR55.43(b)(2).		
Additional Information	This is a high cog question written at the analysis/comprehension level. The examinee has to analyze the plant conditions provided in the stem, determine the effect on the APRM channels and the operability of those APRM channels based on that analysis (3-SPK).		
NRC Exams Only			
Question Type	Bank (CL-ILT-A12089)	**Difficulty**	N/A
Technical Reference and Revision #	ITS B3.3.1.1 (B3.3-7) Rev. 7-5		
Training Objective	215005.14 Given AVERAGE POWER RANGE MONITOR System operability status and a copy of Tech Specs, DISCUSS the bases for the AVERAGE POWER RANGE MONITOR System Tech Spec LCO, related safety limits and Limiting Safety System Settings		
Previous NRC Exam Use	None		

13	ID: 2096893	Points: 1.00

The plant was performing a heatup and pressurization with reactor pressure at 175 psig when annunciator 5050-1C NOT AVAILABLE VY SYSTEM DIVISION 1 was received due to a trip of 1VY04C RCIC PMP RM SPLY FAN.

Which of the following actions, if any, are required by Technical Specifications?

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: [redacted]

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to RCIC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE. AND A.2 Restore RCIC System to OPERABLE status.	[redacted] 14 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Reduce reactor steam dome pressure to ≤ [redacted] psig.	12 hours 36 hours

A.	No action is required. The reactor startup may continue.

B.	Verify HPCS is operable. Entry into Mode 1 is NOT allowed until 1VY04C is repaired and RCIC operability is restored.

C.	Verify HPCS is operable. Lower reactor pressure to ≤ the LCO applicability value until 1VY04C is repaired and RCIC operability is restored.

D.	Verify HPCS is operable. Entry into Mode 1 is allowed. Operation in Mode 1 or 2 may continue for up to 14 days. If RCIC is not operable in 14 days then be in Mode 3 in 12 hours and reduce RPV pressure ≤ the LCO applicability value in 36 hours.

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per ITS 3.5.3, RCIC is required to be operable in Mode 1, and Modes 2 and 3 with reactor steam dome pressure > 150 psig. Since the stem states that a heatup and pressurization (Mode 2) is in progress and reactor steam dome pressure is 175 psig, then RCIC is required to be operable.   Per ITS 1.0 Use and Application, OPERABILITY is defined as: "A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and **when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).**   Since the room cooling fan is not capable of performing its related support function, the RCIC system is INOPERABLE and the actions of ITS 3.5.3 must be taken.   Per ITS 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;   b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (; exceptions to this Specification are stated in the individual Specifications,); or   c. When an allowance is stated in the individual value, parameter, or other Specification.   This specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.   Since the completion time for Required Action A.2 is 14 days and not an unlimited period of time, then entry into Mode 1 is not permitted until RCIC is restored to an operable status.   Incorrect responses: A is incorrect but plausible. This answer would be correct if 1VY04C was not required for RCIC operability.   C is incorrect but plausible. This action could be taken, but is not required by Technical Specifications.   D is incorrect but plausible. This answer would be correct if the plant were operating in Mode 1.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.217000	B2.1.20	4.6	4.6	2	1	N/A

System Name
Reactor Core Isolation Cooling System (RCIC)

Category Statement
Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q13/88 217000 2.1.20

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the candidate must interpret procedure steps based on plant conditions and select the appropriate procedure steps to execute.
SRO-Only Justification	This question is linked to 10CFR55.43(b)(2) - Facility operating limitations in the technical specifications and their bases. Also linked to 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Additional Information	This is a high cog question written at the application level. The examinee has to analyze plant conditions, apply knowledge of plant procedures (Technical Specifications) and using a provided Technical Specification reference, determine the appropriate action to be taken next (3-SPK/R).

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	ITS 3.5.3 Amend. 216 ITS 3.0.4 Amend. 220 ITS 1.0 Amend. 216		
Training Objective	217000.12 Given REACTOR CORE ISOLATION COOLING (RI) System operability status OR key parameter indications, plant conditions, and a copy of Tech Specs, DETERMINE if Tech Spec Limiting Condition for Operations have been met, and required actions if any.		
Previous NRC Exam Use	None		

14	ID: 2105735	Points: 1.00

A plant startup is in progress.

- Reactor coolant temperature is 180°F.
- The first gang of control rods has been withdrawn to notch 48.

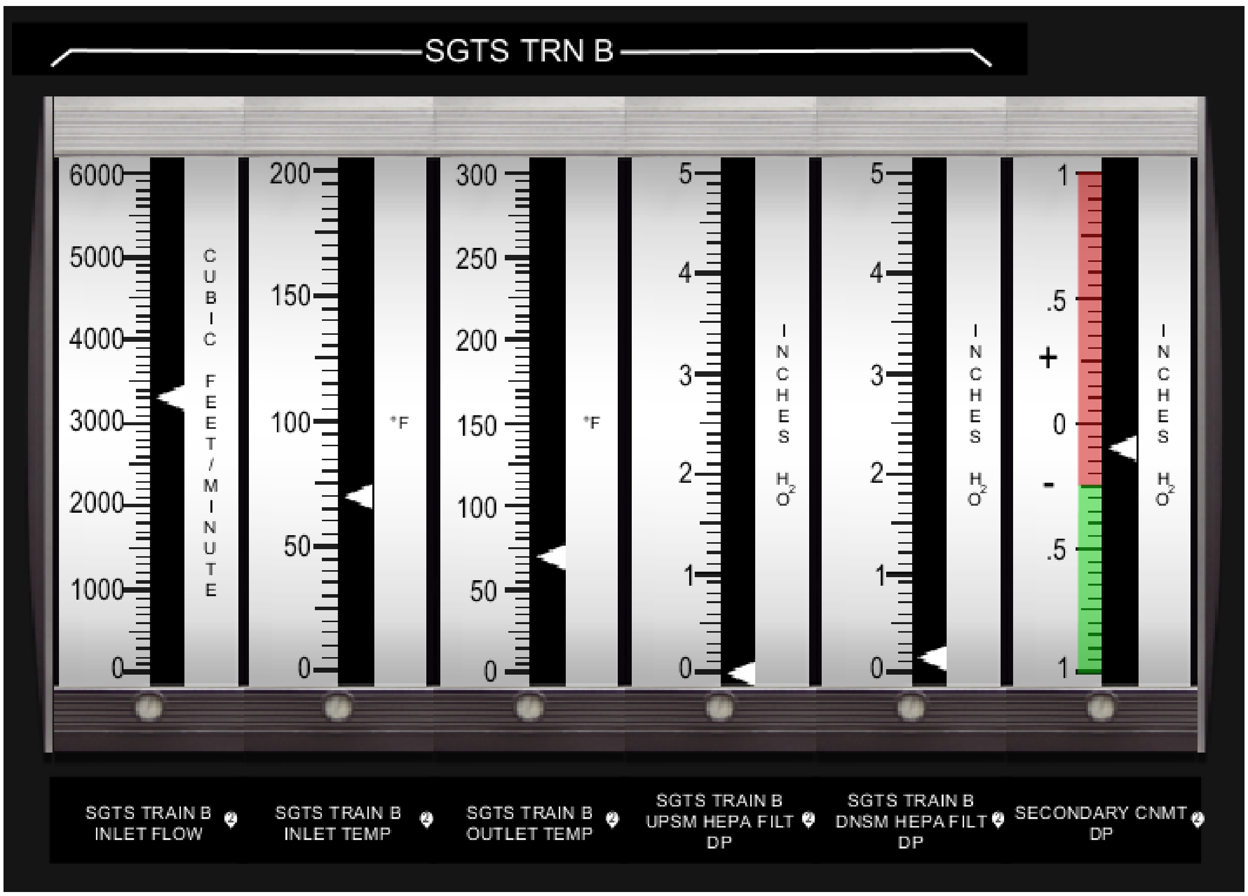
Then, a damper failure caused the Fuel Building Ventilation (VF) fans to trip off on low flow.

- The VF system cannot be restarted.

Then, when starting Standby Gas Treatment System (SGTS) Train A, 0VG02CA SGTS Train A Exhaust Fan breaker tripped.

- SGTS Train B was started successfully.

Thirty (30) minutes later, the following was observed at 1H13-P801 (trend is steady on all indications):



What is the most limiting action required per ITS, if any?

A.	Enter ITS LCO 3.0.3.
B.	No actions are required.
C.	Be in MODE 3 in 12 hours.

D.	Restore SGTS A to OPERABLE in 7 days.

Answer:	C

Answer Justification / Plausibility Statements
C is correct: With a plant startup in progress and control rods withdrawn, the plant is in Mode 2. Per ITS 3.6.4.3, Standby Gas Treatment (SGT) System, two SGT subsystems are required to be operable. The trip of the 'A' VG exhaust fan renders the 'A' train INOPERABLE. An indication of SGTS B inlet flow < 3600 CFM is outside the allowable band of 4000 CFM ± 10% (i.e. 3600 – 4400 CFM) per ITS SR 3.6.4.3.2 and ITS 5.5.7.d. In this situation, SGTS Train B is also INOPERABLE, resulting in both SGTS Trains being INOPERABLE, which necessitates entry into ITS LCO 3.6.4.3 condition D, requiring the plant to be in MODE 3 in 12 hours. Incorrect Responses: A is incorrect but plausible. This answer is plausible because ITS 3.6.4.3 condition D and ITS 3.0.3 both require the plant to be placed in MODE 3. A common misconception exists that any required shutdown statement contained in ITS is equivalent to ITS 3.0.3. B is incorrect but plausible. This answer would be correct if the plant was in MODE 4 (based solely on a reactor coolant temperature of 180°F), in which case no ITS actions would be required. D is incorrect but plausible. This answer would be correct if SGTS Train B was OPERABLE. However, the SGTS Train B is INOPERABLE based on train inlet flow and ITS 3.6.4.3 must be applied to both trains of SGTS.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
261000.A2.05	A2.05	3.0	3.1	2		9

System Name
Standby Gas Treatment System

Category Statement
Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

K/A Statement
Fan trips

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

290001.12	290001.12 Given FUEL BUILDING HVAC AND SECONDARY CONTAINMENT System operability status OR key parameter indications, plant conditions, and a copy of Tech Specs, DETERMINE if Tech Spec Limiting Condition for Operations have been met, and required actions if any.

	Q14/89 261000 A2.05

Other NRC Data

References Provided	ITS LCO 3.6.4.3 (pages 3.6-51 and 52) with the following redactions: • LCO and applicability statement • C.1 and C.2 completion times
K/A Justification	This question satisfies the K/A statement in that the indications are used to predict the state of operability of the SGTS subsystems following a fan trip, and from that, applying the correct LCO required action in order to mitigate the consequences of the malfunctions.
SRO-Only Justification	This question requires the examinee to evaluate the indications presented in a written and visual format, make an Operability determination (an SRO-only function) based on these indications, and select the correct actions to be performed. This question is linked to 10CFR55.43b(2).
Additional Information	This is a high cog question written at the analysis level. The examinee has to analyze several parameters provided in the stem, and then determine required actions based on the analysis (3-SPK/SPR).

NRC Exams Only			
Question Type	Bank (CL-ILT-1649677)	**Difficulty**	N/A
Technical Reference and Revision #	• ITS LCO 3.6.4.3 (3.6-53) Amend. 216 • ITS 5.5.7 (5.0-13) Amend. 95		
Training Objective	261000.12 Given VG SBGT STANDBY GAS TREATMENT System operability status OR key parameter indications, plant conditions, and a copy of Tech Specs, DETERMINE if Tech Spec Limiting Condition for Operations have been met, and required actions if any.		
Previous NRC Exam Use	None		

15	ID: 2105847	Points: 1.00

The reactor was operating at rated thermal power with the upper containment pool gates in their normal online configuration.

THEN, a loss of Service Air/Instrument (SA/IA) Pressure occurred:

- Reactor scrammed, all rods in.
- Main Steam Equalizing header pressure is 0 psig.
- ADS Backup Air Bottle Pressures are 2400 psig each.

SA/IA pressure is now recovering and rising from a low of 40 psig.

Which of the following actions is required to be performed?

A.	Place all SRV control switches to OFF.
B.	SHUT the 1B33-F075A(B) Pmp A(B) Seal Staging Shutoff Valve(s).
C.	Verify the Backup Air Supply is supplying the Reactor Cavity to Steam Dryer Pool Gate Seal.
D.	Place all Main Steam Isolation Valve (MSIV) control switches to CLOSE.

Answer:	D

Answer Justification / Plausibility Statements
D is correct. Per CPS 4004.01 Instrument Air Loss, the control switches for Main Steam Isolation Valves (MSIVs) are required to be placed to CLOSE to prevent an inadvertent reopening when air pressure is restored. Incorrect Responses: A is incorrect but plausible. This answer would be correct in accordance with EOP-1 if all IA to the SRVs was lost, including the back up air bottles. Based on the plant conditions presented in the stem, the ADS Backup Air Bottle Pressures are shown in the stem with normal values. B is incorrect but plausible because 1B33-F075A(B) Pmp A(B) Seal Staging Shutoff Valves are air operated valves, and if they failed closed would result in destaging the RR Pump seals. Taking the control switches to close would then permit restoring seal staging under controlled conditions. This action is incorrect, however, because the seal staging valves fail open on a loss of air. C is incorrect but plausible. This answer would be correct if refueling operations were in progress with the Reactor Cavity to Steam Dryer Pool Gate in use.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.300000	B2.1.32	3.8	4.0	2	1	N/A

System Name
Instrument Air System (IAS)

Category Statement
Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.2	43.2

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q15/90 300000 2.1.32

PB400401.01.04	PB400401.01.04 Describe the effect a loss of Instrument Air has on the following: .1 Reactor Protection System (RPS) .2 Main Steam Isolation Valves .3 Automatic Depressurization System (ADS) .4 Condenser Vacuum System (CA) .5 Feedwater System (FW) .6 Condensate/Condensate Booster System (CD/CB) .7 Reactor Water Cleanup System (RT) .8 Off-Gas System (OG) .9 Drywell Purge System (VQ) .10 CNMT, Fuel, Auxiliary, Turbine and Radwaste Building HVAC systems (VR, VF, VA, VT, VW) .11 Plant Chillers (WO) .12 Drywell and CNMT Equipment/Floor Drain System (RE/RF)

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to analyze the conditions in the stem and then determine the required actions to answer the question. The explanation of system limits and precautions is inferred by choosing the correct procedure actions to perform.
SRO-Only Justification	This question is linked to 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Additional Information	Question is High Cog, written at the analysis and application level. Requires examinee to analyze conditions presented in the stem and determine required actions (2-RI).
NRC Exams Only	
Question Type	Bank (CL-ILT-N11082) **Difficulty** N/A
Technical Reference and Revision #	• CPS 4401.01 Rev. 30 • CPS 3214.01 Rev. 27c • CPS 3302.01 Rev. 37b • CPS 4004.01 Rev. 10a
Training Objective	PB400401.01.04 Describe the effect a loss of Instrument Air has on the following: .2 Main Steam Isolation Valves
Previous NRC Exam Use	CPS ILT 10-1 NRC Exam

16	ID: 2096989	Points: 1.00

The plant was operating at rated thermal power (RTP) when the 'B' Reactor Recirculation (RR) Pump tripped.

CPS 4008.01 Abnormal Reactor Coolant Flow was entered and appropriate immediate operator actions were taken.

Current plant parameters are as follows:

Parameter	Value	Trend
Reactor Power	67%	stable
Core Flow	48.5 mlbh	stable

What actions are required to be taken?

- Action 1 - Enter ITS 3.4.1 Condition A (Recirculation loop jet pump flow mismatch not within limits).
- Action 2 - Reduce thermal power ≤ 58% by reducing reactor recirculation flow within 4 hours.
- Action 3 - Reduce thermal power ≤ 58% by inserting control rods <u>and</u> recirculation flow within 4 hours.

Action...

A.	1 and 2
B.	1 and 3
C.	2 only
D.	3 only

Answer:	D

Answer Justification / Plausibility Statements

D is correct:

The trip of Reactor Recirculation Pump 'B' will result in:

- operation outside the controlled entry region and below the MELLLA limit of the CPS Stability Control & Power/Flow Operating Map (CPS 3005.01 Unit Power Changes Figure 1)
- Single loop operations with power above the ITS 3.4.1 B.1 limit for reactor power in single recirculation loop operation (≤ 58% RTP).

This will require performance of ITS 3.4.1 RA B.1 and CPS 3005.01 step 8.4.2 to reduce thermal power to ≤ 58% RTP by inserting control rods and reducing recirculation flow within 4 hours.

Reducing power to ≤ 58% RTP must be accomplished by inserting control rods and reducing recirculation flow. Reducing power to ≤ 58% RTP with RR flow alone will result in operation in the Controlled Entry Region. Per CPS 3005.01 Limitation 6.4.4, entry into the controlled entry region below the MELLLA limit is only permitted as part of a planned power change (ReMA identified) and is therefore prohibited in this scenario.

Incorrect responses:

A is incorrect but plausible:

- RR loop flows are mismatched due to the trip of RR Pump B. However, per ITS B3.4.1, required action A.1 is only required with both recirculation loops operating but the flows not matched. Since RR Pump B is not operating, ITS 3.4.1 RA A.1 is not required to be taken.
- Action 2 is plausible because ITS 3.4.1 B.1 requires thermal power to be reduced to ≤ 58% RTP. Power reductions are routinely performed using RR flow reduction but is prohibited in this case because voluntary entry into the controlled entry region is prohibited.

B is incorrect but plausible - RR loop flows are mismatched due to the trip of RR Pump B. However, per ITS B3.4.1, required action A.1 is only required with both recirculation loops operating but the flows not matched. Since RR Pump B is not operating, ITS 3.4.1 RA A.1 is not required to be taken.

C is incorrect but plausible because ITS 3.4.1 B.1 requires thermal power to be reduced to ≤ 58% RTP. Power reductions are routinely performed using RR flow reduction but is prohibited in this case because voluntary entry into the controlled entry region is prohibited.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
202002.A2.04	A2.04	3.0	3.2	2		1

EXAMINATION ANSWER KEY

ILT 18-1 NRC SRO Written Exam

System Name
Recirculation Flow Control System

Category Statement
Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

K/A Statement
Recirculation pump speed mismatch between loops: Plant-Specific

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q16/91 202002 A2.04

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the candidate must select the correct impact of the abnormal condition presented (Recirculation loop jet pump speed/flow mismatch not within limits) and then select the correct response to mitigate the abnormal condition.		
SRO-Only Justification	This question is linked to SRO only task 999999.07 Apply Technical Specification requirements and 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases. Specifically, the question involves technical specifications and the application of technical specifications to mitigate the abnormal condition.		
Additional Information	This is a high cog question written at the analysis and application level. The examinee has to analyze plant conditions following an event and then apply knowledge to determine the correct actions to mitigate the event.		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• ITS 3.4.1 (3.4-1 & 3.4-2) Amend. 171 • ITS B3.4.1 (B3.4-4) Rev. 11-1 • CPS 3005.01 Rev. 43f		
Training Objective	202002.14 Given Reactor Recirculation System operability status and a copy of Tech Specs, DISCUSS the bases for the Reactor Recirculation System Tech Spec LCO, related safety limits and Limiting Safety System Settings.		
Previous NRC Exam Use	None		

17	ID: 2106217	Points: 1.00

The unit is operating at rated thermal power.

- Div 1 DG is out of service (both air receivers are depressurized).
- Appropriate LCOs have been entered.

THEN:

- Annunciator 5065-8B RHR B Out Of Service was received due to a blown pump motor breaker close control power fuse.

The conditions above remain unchanged for 8 hours.

Which of the listed LCO Conditions must be entered during this 8 hour time period?

	LCO	Condition
1	3.6.2.4	C. One SPMU subsystem inoperable for reasons other than Condition A or B (upper containment pool and suppression pool water levels NOT within limits or upper containment pool water temperature NOT within limit).
2	3.6.1.9	B. Two Feedwater Leakage Control Systems (FWLCS) inoperable.
3	3.5.1	H. Three or more ECCS injection / spray subsystems inoperable.

A.	1 ONLY
B.	2 ONLY
C.	2 AND 3 ONLY
D.	1, 2, AND 3

Answer:	C

Answer Justification / Plausibility Statements
C is correct: Per OP-CL-108-104-1001 ITS LCO/ORM OR/ODCM OR Evaluations And Guidance For Safety Function Determination, step 2.6: If a diesel generator becomes inoperable, declaring a **supported system** inoperable is not required unless the **supported system** in the redundant division is inoperable, as required by LCO 3.8.1, Required Action B.2 or C.1. When a diesel generator becomes inoperable in MODE 1, 2, or 3, ITS LCO 3.8.1, AC Sources - Operating, Required Action B.2 requires identification of inoperable required features which are redundant to required features supported by the inoperable diesel generator. With the Div 1 DG inop and RHR 'B' inop, LCO Required Action 3.8.1 B.2 must be entered (Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable). The following LCO conditions must be entered: • ITS 3.6.1.7 Condition B - Two RHR Containment Spray subsystems inoperable • ITS 3.6.1.9 Condition B - Two Feedwater Leakage Control subsystems inoperable • ITS 3.6.2.3 Condition C - Two RHR suppression pool cooling subsystems inoperable • ITS 3.5.1 Condition H - Three or more ECCS injection / spray subsystems inoperable (LPCS, RHR 'A', and RHR 'B') Incorrect Responses: A is incorrect but plausible due to an incorrect interpretation of the support-supported relationship between RHR and SPMU. B is incorrect but plausible. The answer is partially correct in that 3.6.1.9 B must be entered. It would be plausible to exclude 3.5.1 H if an incorrect interpretation of "redundant required features supported by the inoperable diesel generator" excluded LPCS, such that only 2 injection/spray subsystems were deemed inoperable (RHR 'A' and 'B'). D is incorrect but plausible and is partially correct (2 and 3 are required to be entered) due to an incorrect interpretation of the support-supported relationship between RHR and SPMU.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.223001	B2.2.40	3.4	4.7	2	2	N/A

System Name
Primary Containment System and Auxiliaries

Category Statement
Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.2	43.2
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

264000.12	264000.12 Given DIESEL GENERATOR/DIESEL FUEL OIL System operability status OR key parameter indications, plant conditions, and a copy of Tech Specs, DETERMINE if Tech Spec Limiting Condition for Operations have been met, and required actions if any. .1 SX .2 DC Electrical Distribution (DC) .3 AC Electrical Distribution (AP) .4 Diesel Generator Ventilation (VD)

264000.12	264000.12 Given DIESEL GENERATOR/DIESEL FUEL OIL System operability status OR key parameter indications, plant conditions, and a copy of Tech Specs, DETERMINE if Tech Spec Limiting Condition for Operations have been met, and required actions if any. .1 SX .2 DC Electrical Distribution (DC) .3 AC Electrical Distribution (AP) .4 Diesel Generator Ventilation (VD)

	Q17/92 223001 2.2.40

Other NRC Data

References Provided	ITS 3.8.1 (3.8-1, 3.8-2) with the following redactions: • LCO and applicability statements • 1 hour or less completion time statements
K/A Justification	This question requires the examinee to analyze the conditions listed in the stem and determine the appropriate TS conditions that must be entered.
SRO-Only Justification	This question is linked to 10CFR55.43(b)(2) - Facility operating limitations in the technical specifications and their bases.
Additional Information	This question is high cog, written at the analysis/comprehension level. The examinee has to analyze the conditions provided in the stem and then determine the LCO(s) that must be entered based on that analysis (3-SPK).

NRC Exams Only

Question Type	Bank (CL-ILT-A12096)	**Difficulty**	N/A
Technical Reference and Revision #	• ITS 3.8.1, (3.8-2) Amend.141 • OP-CL-108-104-1001, Rev. 11a		
Training Objective	223001.12 Given PRIMARY CONTAINMENT System operability status OR key parameter indications, plant conditions, and a copy of Tech Specs, DETERMINE if Tech Spec Limiting Condition for Operations have been met, and required actions if any.		
Previous NRC Exam Use	None		

18	ID: 2106208	Points: 1.00

A plant heatup and pressurization is in progress. Reactor pressure is 50 psig.

THEN, a HIGH alarm is received (NOT a spike) on the Noble Gas channel for 0RIX-PR001, HVAC EXHAUST PRM #1.

NO system actuations occurred.

Which of the following actions is required?

A.	Secure the Steam Jet Air Ejector.
B.	Declare 0RIX-PR001 non-functional.
C.	Stop the Mechanical Vacuum Pump.
D.	Shift Main Control Room Ventilation (VC) to High Rad mode.

Answer:	C

Answer Justification / Plausibility Statements
C is correct. Per CPS 4979.01 Abnormal Release of Airborne Radioactivity, AR/PR Action Table 2: **IF** High Alarm [5140.41 / 42] due to: 0RIX-PR001/2 HVAC Exhaust PRM #1 / #2 **THEN** Stop any running Vacuum Pump(s), 0CA01PA (B) (to secure the release pathway). Based on plant conditions, the CRS should direct the appropriate Reactor Operator to secure the running Vacuum Pump. Incorrect Responses: A is incorrect but plausible. The Steam Jet Air Ejector (SJAE) draws air and non-condensibles from the main condenser like the mechanical vacuum pumps, however the SJAEs are not placed in service until main steam pressure is at or above 200 psig. Additionally, the SJAEs discharge to the Off Gas system and 0RIX-PR001 does not monitor Off Gas effluent to the stack. B is incorrect but plausible. This answer would be correct if the monitor provided system actuations that did not occur or if multiple PRMs monitored HVAC exhaust. 0RIX-PR001 and 002 monitor HVAC exhaust but only one monitor is in service at a time. D is incorrect but plausible. This answer would be correct if a High Alarm on 0RIX-PR001 HVAC Exhaust PRM #1 required a shift of VC to high rad mode. However, per CPS 4979.01 Abnormal Release of Airborne Radioactivity, AR/PR Action Table 2: **IF** High Alarm [5140.64] due to: 1RIX-PR009A-D, MCR Air Intake Ducts, **THEN** Refer to CPS 3402.01 Control Room HVAC (VC), High Radiation Isolation section, and is therefore not applicable to the current plant conditions.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
272000.A2.11	A2.11	3.4	3.7	2		7

System Name
Radiation Monitoring System

Category Statement
Ability to (d) predict the impacts of the following on the RADIATION MONITORING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

K/A Statement
Leakage and/or breaks from contaminated systems to atmosphere or to other process systems

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.5	41.5

10CFR55-43b (SRO) Data

Para Num	Text
N/A	N/A

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q18/93 272000 A2.11

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because it requires the examinee to determine which procedure is used to correct, control, or mitigate the effects of a PRM going into High Alarm due to leakage from a contaminated system into a process system.
SRO-Only Justification	This question is linked to 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency conditions.
Additional Information	Question is Low Cog, written at the memory level. Requires recall of procedure steps (1-P).

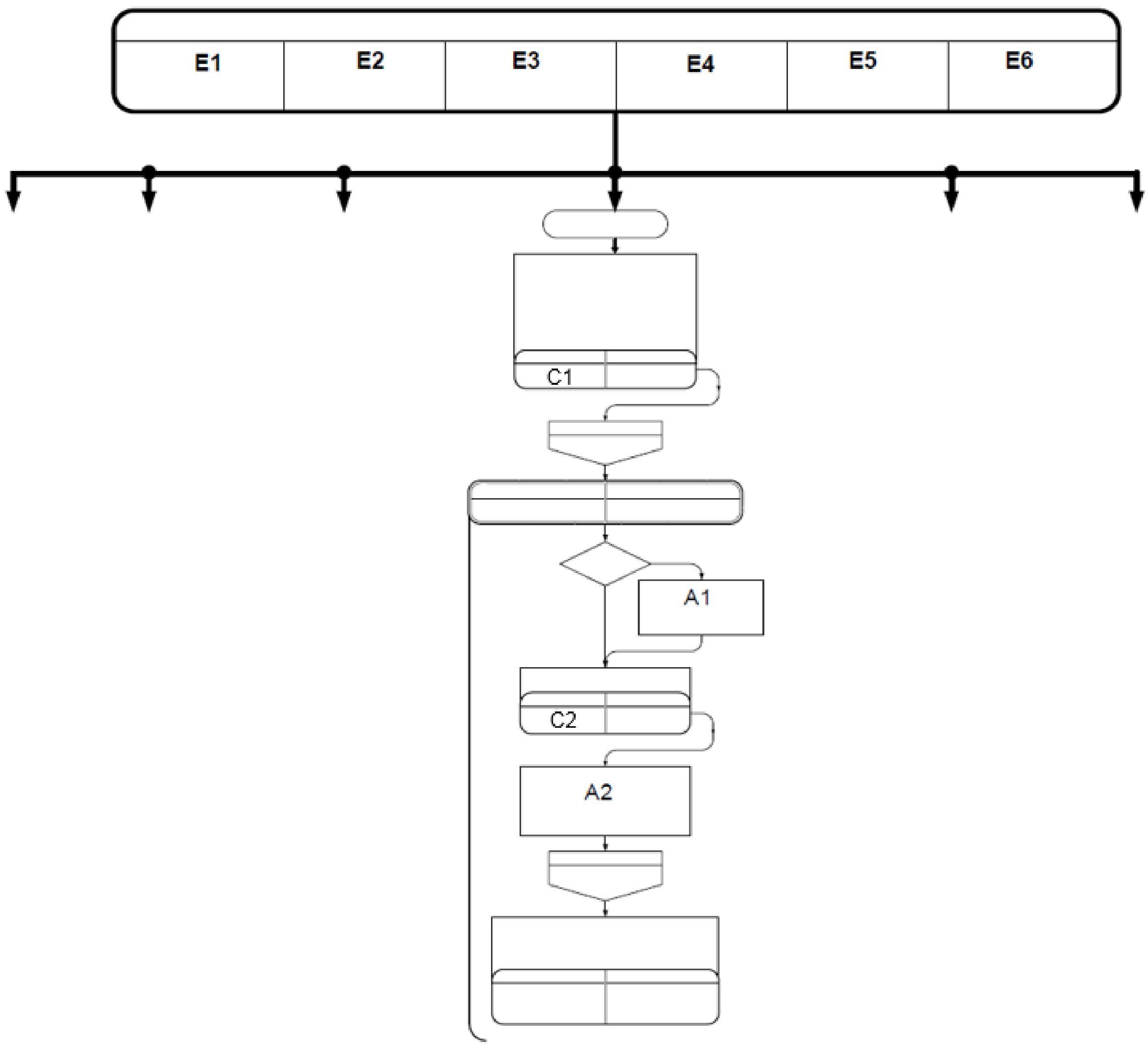
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3315.03 Rev. 10 • CPS 3402.01 Rev. 32c • CPS 4979.01 Rev. 10e • CPS 5140.41 Rev. 1g		
Training Objective	DB497901.01.04 Given a set of conditions, describe actions to be taken for radiation monitor alarms associated with CPS No. 4979.01, ABNORMAL RELEASE OF AIRBORNE RADIOACTIVITY.		
Previous NRC Exam Use	None		

19	ID: 2106204	Points: 1.00

The plant was operating at rated thermal power when a transient occurred requiring entry into the generic flowchart shown below.

Part 1 - An arrow pointing to Condition C1 indicates that the crew ____(1)____.

Part 2 - The flowchart has been completed down to A1 but A1 actions have NOT yet been taken. If C2 conditions are exceeded, the CRS will direct the Reactor Operators to perform actions of _____(2)_____ NEXT.



A.	(1) has completed the referenced step (2) A1

B.	(1) has completed the referenced step (2) A2

C.	(1) is maintaining or waiting for a specific plant condition (2) A1
D.	(1) is maintaining or waiting for a specific plant condition (2) A2

Answer:	C

Answer Justification / Plausibility Statements

C is correct:

Per CPS 1005.09 Emergency Operating Procedure (EOP) and Severe Accident Guideline (SAG) Program, step 8.12.4 EOP Performance:

> The following items represent the expected actions to be taken by the operating crew during actual use of the EOPs:
>
> > 5. When actions have progressed to a point where the crew is maintaining or waiting for specific plant condition, an arrow should be used to mark their place.

Per OP-CL-101-111-1001 Strategies For Successful Transient Mitigation, 4.1.3 EOP Execution, step 4:

> **All steps must be executed in their specified order** when executing control legs of the EOPs. This is to ensure all available mitigating systems are utilized and their effectiveness assessed, even if a blowdown parameter is currently exceeded.

Incorrect Responses:

A is incorrect but plausible. CPS 1005.09 requires steps that are in progress (NOT completed) to be annotated when the user proceeds to the next subsequent step, however an arrow is used to indicate that the crew is waiting for a specific plant condition prior to proceeding.

B is incorrect but plausible.

- CPS 1005.09 requires steps that are in progress (NOT completed) to be annotated when the user proceeds to the next subsequent step, however an arrow is used to indicate that the crew is waiting for a specific plant condition prior to proceeding, and
- if EOP actions were required to be performed when a condition parameter is exceeded regardless of the status of the preceding EOP actions.

D is incorrect but plausible. This answer would be correct if EOP actions were required to be performed when a condition parameter is exceeded regardless of the status of the preceding EOP actions.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.1.06	B2.1.06	3.8*	4.8	3	N/A	N/A

System Name
Conduct of Operations

Category Statement
Ability to manage the control room crew during plant transients. (CFR: 41.10 / 43.5 / 45.12 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q19/94 2.1.6

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to demonstrate the ability to manage the control room crew by selecting the actions they are required to perform during execution of EOPs to answer the question.		
SRO-Only Justification	This question is linked to SRO only task 100509.07 Execute EOP Decision Symbols.		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall the procedural actions required to execute the EOPs (1-P).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 1005.09 Rev. 11 • OP-CL-101-111-1001 Rev. 15b		
Training Objective	LP85801.2.1.6 Ability to manage the control room crew during plant transients		
Previous NRC Exam Use	None		

20	ID: 2107809	Points: 1.00

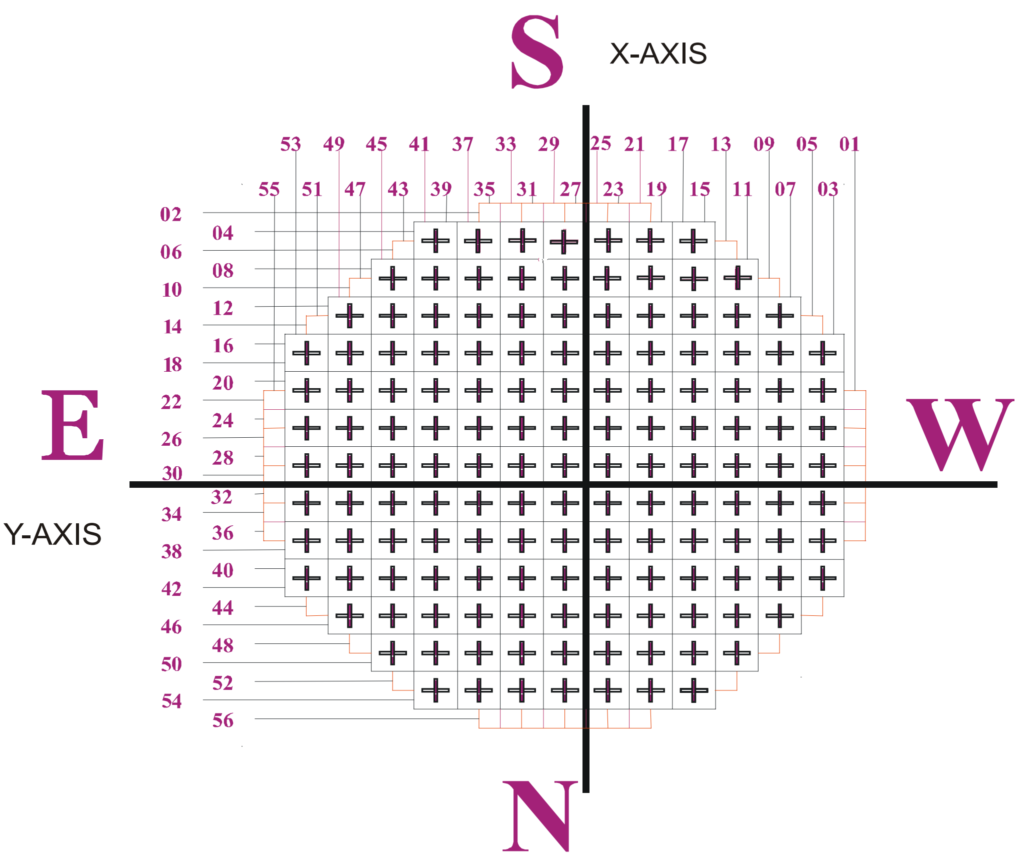
Refueling operations are in progress.

A fuel assembly was inserted into core location 37-12 when it was discovered that the bundle was mis-oriented.

- The slack cable light is lit
- The grapple has NOT been released.

The fuel bundle mis-orientation _____(1)____ be corrected without Reactor Engineer guidance.

The correct orientation is with the channel fastener pointing _____(2)_____.



A.	(1) can (2) SE

B.	(1) can (2) NW

C.	(1) can NOT (2) SE

D.	(1) can NOT (2) NW

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Part 1 Per CPS 3703.01 Core Alterations, Limitation 6.24 Fuel Bundle Handling, step 6.24.1 states that if a fuel bundle is placed in an incorrect core location OR mis-oriented in the correct location and has not been released, then remove the bundle from the incorrect core location and place it in the correct core location as per the approved SNM Move sheets. Initiate an Issue Report to document this condition and notify a Reactor Engineer for evaluation of impact on Shutdown Margin during fuel shuffle. Part 2 Per NF-AA-330-1001 Core Verification Guideline, step 4.2.10 and Attachment 4, the fuel assembly is correctly oriented when the fuel assembly spring clip or channel fastener is located at the wide-wide (control rod) corner. Since fuel assembly 37-12 is located at the intersection of 37 on the X-axis, and 12 on the Y-axis, in the top left of the fuel cell (SE), the channel fastener should point toward the center of the cell (NW). Incorrect Responses: A is incorrect but plausible. Part 1 is correct. Part 2 is plausible if the coordinates for control rod 12-37 is used. Control rod 12-37 is located in the NE quadrant of the core. Correct orientation for a fuel bundle in the lower right of the NW quadrant is SE. C is incorrect but plausible: • Part 1 would be correct if the grapple had been released. CPS 3703.01 step 6.24.2 states that a Reactor Engineer must provide further guidance for correcting the mis-orientation once the grapple has been released. • Part 2 is plausible if the coordinates for control rod 12-37 is used. Control rod 12-37 is located in the NE quadrant of the core. Correct orientation for a fuel bundle in the lower right of the NW quadrant is SE. D is incorrect but plausible. Part 1 would be correct if the grapple had been released. CPS 3703.01 step 6.24.2 states that a Reactor Engineer must provide further guidance for correcting the mis-orientation once the grapple has been released.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.1.36	B2.1.36	3.0	4.1	3	N/A	N/A

System Name

Category Statement
Knowledge of procedures and limitations involved in core alterations. (CFR: 41.10 / 43.6 / 45.7)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.6	43.6

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q20/95 2.1.36

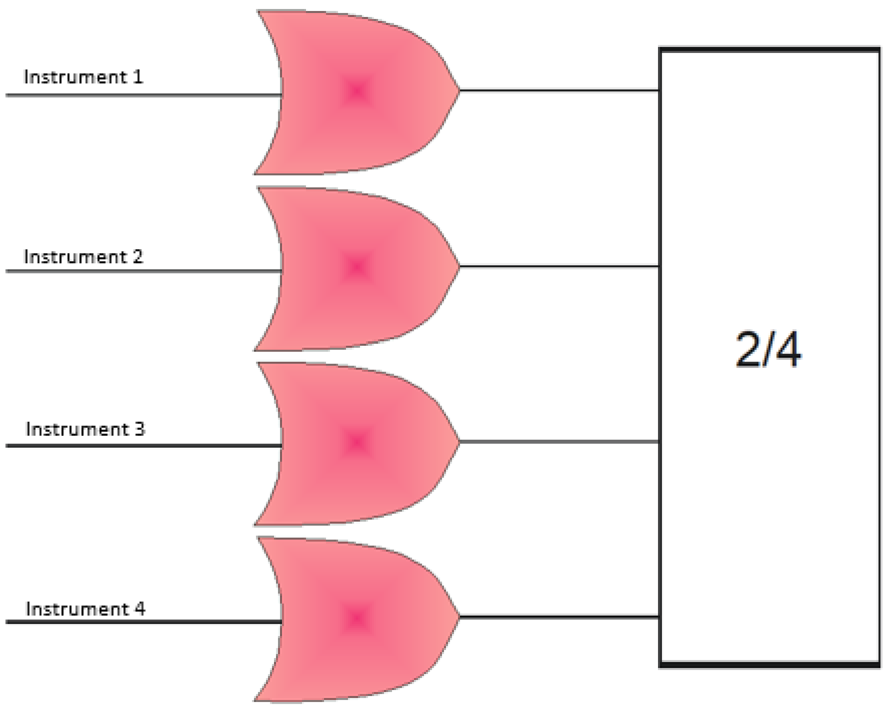
Other NRC Data

References Provided	Yes - Internal		
K/A Justification	This question meets the KA because has to demonstrate knowledge of procedures and limitations involved in core alterations with regard to incorrect fuel bundle orientation to answer the question.		
SRO-Only Justification	This question is an SRO-only question because the refueling SRO is responsible for ensuring correct orientation of the fuel bundle before releasing the grapple (CPS 3703.01 step 4.22). Also linked to 10CFR55.43(b)(7) Fuel handling facilities and procedures.		
Additional Information	Cog level justification - this question is a high cog question because the examinee has to solve a problem (correct fuel bundle orientation) using knowledge (fuel bundle construction, relationship of the fuel bundle to the fuel cell, etc.).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• CPS 3703.01 Rev. 28b • NF-AA-330-1001 Rev. 12		
Training Objective	LP86610.01.09 Describe four (4) methods used to verify proper fuel assembly orientation.		
Previous NRC Exam Use	None		

21	ID: 2108183	Points: 1.00

The plant was operating at rated thermal power.

THEN, Instrument 1 failed downscale.



Instrument 1 was tripped as required by Technical Specifications.

ITS 3.0.5 _____(1)_____ permit Instrument 1 to be taken out of trip under admin controls to perform surveillance testing on Instruments 2, 3, and 4.

ITS 3.0.5 _____(2)_____ permit Instrument 1 to be taken out of trip under admin controls to perform troubleshooting on Instrument 1.

A.	(1) does (2) does
B.	(1) does (2) does NOT
C.	(1) does NOT (2) does
D.	(1) does NOT (2) does NOT

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per ITS 3.0.5, equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control **solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment**. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY. Part 1 - ITS 3.0.5 will permit surveillance testing to be performed by returning the inoperable channel to service (removing the trip) under admin controls to demonstrate the operability of the remaining instruments. Part 2 - ITS 3.0.5, however, provides <u>no</u> provision for returning the inoperable instrument to service to perform troubleshooting or corrective maintenance on the inoperable instrument. This would be antithetical to 3.0.5 which states that equipment removed from service may be returned to service under admin controls <u>solely</u> to perform testing required to demonstrate its operability or the operability of other equipment. Incorrect Responses: A is incorrect but plausible. Part 1 is correct. Part 2 is plausible because ITS 3.0.5 will permit inoperable instruments to be returned to service under very specific conditions. C is incorrect but plausible because ITS 3.0.5 provides very specific conditions for returning inoperable instruments to service. D is incorrect but plausible. Part 1 is plausible because ITS 3.0.5 provides very specific conditions for returning inoperable instruments to service. Part 2 is correct.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.2.21	B2.2.21	2.9	4.1	3	N/A	N/A

System Name
Equipment Control

Category Statement
Knowledge of pre- and post-maintenance operability requirements. (CFR: 41.10 / 43.2)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.2	43.2

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q21/96 2.2.21

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to have knowledge of pre- and post-maintenance operability requirements to answer the question. ITS 3.0.5 provides permissible actions to take when operating with inoperable equipment and when restoring inoperable equipment to service.		
SRO-Only Justification	This question is linked to SRO-only task 140109.23 (Apply the administrative requirements for execution of Technical Specifications and Off-Site Dose Calculation Manual Requirements). Also linked to 10CFR55.43(b)(2), Facility operating limitations in the Technical Specifications and their bases.		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall facts from a procedure to answer the question (1-F).		
NRC Exams Only			
Question Type	New	**Difficulty**	N/A
Technical Reference and Revision #	• ITS 3.0.5 (pg. 3.0-2) Amendment No. 213 • ITS SR 3.0.1 (pg. 3.0-4) Amendment No. 213		
Training Objective	LP85802.2.2.21 Knowledge of pre and post maintenance operability requirements.		
Previous NRC Exam Use	None		

22	ID: 2106168	Points: 1.00

Which of the following activities is controlled as a temporary configuration change (NOT a controlled exclusion) per CC-AA-112, Temporary Configuration Changes?

A.	Installation of a jumper in accordance with an approved surveillance test procedure.
B.	Attaching a hose to a system vent fitting in accordance with an approved procedure.
C.	Changing a Control Room alarm setpoint in accordance with an approved troubleshooting plan.
D.	Installing a catch containment under a leaking component in the SX 'A' pump room in accordance with an approved Radiation Protection procedure.

Answer:	C

Answer Justification / Plausibility Statements
C is correct. Per CC-AA-112 Temporary Configuration Changes: 4.2 <u>Controlled Exclusions</u> **DETERMINE** if the activity is considered a Controlled Exclusion from this procedure. **If** the activity is a Controlled Exclusion, **then TAG** and **CONTROL** the change per the appropriate station procedure(s) for implementing the change. 4.2.12 The following activities are typically **<u>not</u>** considered Controlled Exclusions from this procedure. • Temporary Setpoint Changes Per MA-AA-716-004 Conduct of Troubleshooting: 4.1.10.4. Use CC-AA-112 Temporary Configuration Changes, if as-left configuration will be different from plant design 4.2.7. **DETERMINE** whether temporary configuration change control is required in accordance with CC-AA-112, Temporary Configuration Changes, for alterations made during the troubleshooting process. Therefore, changing a Control Room alarm setpoint in accordance with an approved troubleshooting plan is required to be tagged and controlled per CC-AA-112 Temporary Configuration Changes. Incorrect Responses: A is incorrect but plausible. Jumper installation is a temporary configuration change but is considered a controlled exclusion by CC-AA-112 because jumper installation and removals is controlled by its respective surveillance procedure. B is incorrect but plausible. Hose installation is a temporary configuration change but is considered a controlled exclusion by CC-AA-112 because hose installation and removals is controlled by its respective procedure. D is incorrect but plausible. Catch Basin installation is a temporary configuration change but is considered a controlled exclusion by CC-AA-112 because Catch Basins are controlled by Radiation Protection procedure RP-AA-502.

<u>K/A Data</u>

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.2.11	B2.2.11	2.3	3.3	3	N/A	N/A

System Name
Equipment Control

Category Statement
Knowledge of the process for controlling temporary design changes. (CFR: 41.10 / 43.3 / 45.13)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.3	43.3

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q22/97 2.2.11

Other NRC Data

References Provided	None		
K/A Justification	This question meets the KA because the examinee has to have knowledge of the Temporary Configuration Change process to answer the question.		
SRO-Only Justification	This question is linked to SRO only task 999999.10 CC-AA-112 - Authorize installation of a Temporary Modification. Question is linked to 10CFR55.43(b)(3) Facility licensee procedures required to obtain authority for design and operating changes in the facility.		
Additional Information	This is a low cog question written at the memory level. The examinee has to recall facts from a procedure to answer the question (1-F/1-P).		
NRC Exams Only			
Question Type	Bank (CL-ILT-N15097)	**Difficulty**	N/A
Technical Reference and Revision #	• CC-AA-112, Rev. 27 • MA-AA-716-004, Rev. 16		
Training Objective	LP85802.2.2.5 Knowledge of the process for making design or operating changes to the facility.		
Previous NRC Exam Use	ILT 15-1 NRC Exam		

23	ID: 2105728	Points: 1.00

A fuel damaging LOCA has occurred.

- Containment radiation levels are currently at 43 Rem/hr.
- The TSC has **NOT** been activated.
- The MCR has determined that RPV level restoration will be attempted using the Standby Liquid Control (SLC) Storage Tank IAW CPS 4411.03 Injection Flooding Sources Appendix D: SLC (test or storage tank).

The exposure limit (TEDE) for each person performing local operations on the SLC system is _____(1)_____.

The Shift Manager _____(2)_____ authorize the use of Potassium Iodide (KI) to the personnel assigned to perform local operations on the SLC system.

A.	(1) 10 Rem (2) may
B.	(1) 25 Rem (2) may
C.	(1) 10 Rem (2) may NOT
D.	(1) 25 Rem (2) may NOT

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per EP-AA-113 Personnel Protective Actions, 3. Responsibilities, the Shift Manager (Shift Emergency Director) shall perform the responsibilities of the Station Emergency Director until relieved. The Station Emergency Director is responsible for the following protective actions: • Authorization for emergency exposure greater than 5 Rem • Authorization for issuance of KI to Exelon Nuclear emergency workers and/or onsite personnel • Direction of Assembly, Accountability and Evacuation of personnel. Per EP-AA-113 Attachment 1, the dose limit for protecting valuable property is 10 Rem TEDE when a lower dose is not practical. Per EP-AA-113 section 4.4 KI Assessment, step 4.4.1.B, if workers will be entering an unknown radiological atmosphere that is suspected to have a high iodine concentration (i.e. loss of Fuel Clad barrier) the Shift Manager should recommend the issuance of one (1) 130 mg KI tablet to each emergency worker affected per day for 10 consecutive days. Per EP-AA-1003 Addendum 3 page CL 2-3 Hot Matrix (barrier criteria), a containment radiation monitoring reading > 41.3 R/hr represents a loss of the Fuel Clad. Incorrect Responses: B is incorrect but plausible. 25 rem is the emergency exposure limit for lifesaving operations. The second part of the question is correct. C is incorrect but plausible. The first part of the question is correct. The Shift Emergency Director is responsible for issuing KI to emergency workers and/or onsite personnel. This answer would be correct if the Shift Manager had been relieved as the Station Emergency Director or if a loss of fuel clad barrier had not occurred. D is incorrect but plausible - 25 rem is the emergency exposure limit for lifesaving operations, and the Shift Emergency Director is responsible for issuing KI to emergency workers and/or onsite personnel. This answer would be correct if the Shift Manager had been relieved as the Station Emergency Director or if a loss of fuel clad barrier had not occurred.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.3.14	B2.3.14	3.4	3.8	3	N/A	N/A

System Name

Category Statement
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.12	41.12

10CFR55-43b (SRO) Data

Para Num	Text
43.4	43.4

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q23/98 2.3.14

Other NRC Data

References Provided	None
K/A Justification	This question meets the KA because the examinee has to have knowledge of the emergency exposure hazards to answer the question.
SRO-Only Justification	This question is linked to SRO only task 997777.03 Emergency Plan Activities performed by an SRO. The Station Emergency Director position is filled by the Shift Manager prior to transferring command and control to the Station Emergency Director.
Additional Information	Question is Low Cog, written at the memory level. Requires recall of procedure steps (1-B).
NRC Exams Only	
Question Type	New; **Difficulty** N/A
Technical Reference and Revision #	• EP-AA-113 Rev. 13 • EP-AA-1003 Addendum 3 Rev. 2
Training Objective	LP85803.2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
Previous NRC Exam Use	N/A

24	ID: 2106143	Points: 1.00

0100 - The plant was operating at rated thermal power when a lightning strike resulted in the following:

- A fire in the screenhouse reported by a field operator.
- High alarm on 1RIX-AR035 Main Control Room ARM.
- Loss of the following communication systems
    - Plant radios
    - Gaitronics
    - ENS phones
    - Satellite Phones
    - HPN
    - All Telephone Lines

RP survey in the vicinity of 1RIX-AR035 is reading 0.1 mrem/hr.

0120 - The screenhouse fire is out. Damage is limited to 1CW01FA, Traveling Screen 1A.

Which of the following EALs should be declared and why?

A.	Alert due to the fire.
B.	Unusual Event due to the fire.
C.	Alert due to the alarm on 1RIX-AR035.
D.	Unusual Event due to the loss of communications.

Answer:	B

Answer Justification / Plausibility Statements
B is correct. Per EP-AA-1003 Addendum 3 Emergency Action Levels for Clinton Station, Emergency Action Level HU3 (FIRE potentially degrading the level of safety of the plant) and the accompanying HU3 bases: • There has been a fire in ANY Table H2 area (Screenhouse) AND • The fire is not extinguished in < 15-minutes of a FIRE detection indication: • Report from the field (i.e. visual observation) Incorrect Responses: A is incorrect but plausible. This answer would be correct if the MA5 threshold was met due to fire in an H2 area > 15 minutes. The MA5 threshold also requires indications of degraded performance of at least one train of a SAFETY SYSTEM or VISABLE DAMAGE to a SAFETY SYSTEM, component or structure required by Technical Specifications. The traveling screens are not required by Technical Specifications. C is incorrect but plausible. This answer would be correct if the RA3 threshold was exceeded by receiving the alarm on 1RIX-AR035. The 1RIX-AR035 alarm was spurious and readings were verified to be 0.1 mrem/hr which is normal. Additionally, the alarm setpoint of 1RIX-AR035 would have to be > 15 mR/hr (actual High setpoint of 1RIX-AR035 is 2.49 mR/hr). D is incorrect but plausible. This answer would be correct if the MU7 threshold had been exceeded due to the loss of communications systems presented in the stem. Table M3 - Communications Capability lists PCS phones as an onsite and offsite system. Since the PCS phones are not included in the list of lost systems, the threshold for MU7 Loss of all On-Site or Off-site communications capabilities has not been exceeded.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.4.30	B2.4.30	2.7	4.1	3	N/A	N/A

System Name
Emergency Procedures /Plan

Category Statement
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)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q24/99 2.4.30

CPS.318254	LP87537.01.10 Given section 3 of EP-AA-1003, Radiological Emergency Plan Annex For Clinton Station, and plant parameters indicative of one or more of the following events, properly classify the emergency. .01 Fission Product Boundary Failure .02 Fuel Damage/Degraded Core .03 Radiological Emergency .04 Abnormal Reactor Coolant Leaks, Temperatures and/or Pressures .05 Steam Line Breaks/Safety Relief Valve Failure .06 Loss of Shutdown Systems .07 Reactor Scram .08 Electrical Power Failure .09 Control Room Events .10 Fire .11 Security Events .12 Natural Phenomenon .13 Contaminated Injury .14 Other Hazardous Conditions .15 Loss of Annunciators .16 Failure to meet Tech Spec Action Statement

Other NRC Data

References Provided	EP-AA-1003 Addendum 3 (CL 2-2, CL 2-6, CL 2-7, CL 2-9)		
K/A Justification	This question meets the KA because it requires knowledge of events related to system status that must be reported to the NRC during an emergency plant event.		
SRO-Only Justification	This question requires the examinee to have knowledge of the Emergency Plan Classification process and their associated notification requirements which is an SRO only function.		
Additional Information	Question is High Cog, written at the analysis and application level. The examinee has to analyze the conditions provided in the stem and apply references to determine which emergency classification is appropriate based on that analysis (3-SPR).		
NRC Exams Only			
Question Type	Modified (CL-ILT-A12082) Changed location of the fire in the original question to make distractor B the correct answer.	**Difficulty**	N/A
Technical Reference and Revision #	EP-AA-1003 Addendum 3 Rev. 2		
Training Objective	LP87537.01.10 Given section 3 of EP-AA-1003, Radiological Emergency Plan Annex For Clinton Station, and plant parameters indicative of one or more of the following events, properly classify the emergency. .10 Fire		
Previous NRC Exam Use	N/A		

25	ID: 2103888	Points: 1.00

Due to a transient, the Shift Emergency Director (SED) has declared the following:

Time	Declaration
0100	FS1 - due to loss of Fuel Clad (FC) and Reactor Coolant System (RC).
0200	FG1 - due to rising containment radiation levels.

- NO hostile actions are in progress.
- Command and control has NOT been transferred.

Protective Action Recommendations (PARs) are required to be provided to the state no later than _____(1)_____.

Evacuations must be recommended for EP-AA-111-F-07 Clinton PAR Flowchart _____(2)_____ Sub Area(s).

A.	(1) 0115 (2) Table 1

B.	(1) 0115 (2) Table 3

C.	(1) 0215 (2) Table 1

D.	(1) 0215 (2) Table 3

Answer:	D

Answer Justification / Plausibility Statements

D is correct:

Part 1

Per EP-AA-111 Emergency Classification and Protective Action Recommendations step 2.11.2, PARs must be provided the State, and designated local agencies as applicable, within 15 minutes of (1) classification of the General Emergency or (2) any change in recommended actions.

Since the general emergency was declared at 0200 (FG1), PARs must be provided to the state no later than 0215.

Part 2

Per EP-AA-111-F-07 Clinton PAR flowchart, the recommendation is derived as follows:

- Classification is a General Emergency - yes
- Is this the initial PAR? - yes
- Is there a loss of Primary Containment per the EALs? - no - the stem states that FG1 was declared due to rising containment radiation levels. Per barrier matrix for CT - Containment, rising containment radiation levels are listed in the Potential Loss column, so a loss of Containment can be ruled out.
- Is there a Hostile Action event in progress? - no (provided in the stem).
- Is this PAR being made from the Control Room? - yes - the stem states that the Shift Emergency Director has declared FG1 and that command and control has not been transferred.
- Evacuate per Table 3

Incorrect responses:

A is incorrect but plausible:

- Part 1 would be correct if PARs were required to be made at the Site Area Emergency classification where events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.
- Part 2 would be correct if the event were classified as a Rapidly Progressing Severe Accident due to Containment Radiation levels above 97 R/hr. Incorrect because the containment barrier has not been lost.

B is incorrect but plausible. Part 1 would be correct if PARs were required to be made at the Site Area Emergency classification where events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Part 2 is correct.

C is incorrect but plausible. Part 1 is correct. Part 2 would be correct if the event were classified as a Rapidly Progressing Severe Accident due to Containment Radiation levels above 97 R/hr. Incorrect because the containment barrier has not been lost.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.4.44	B2.4.44	2.4	4.4	3	N/A	N/A

System Name
Emergency Procedures /Plan

Category Statement
Knowledge of emergency plan protective action recommendations. (CFR: 41.10 / 41.12 / 43.5 / 45.11)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.10	41.10
41.12	41.12

10CFR55-43b (SRO) Data

Para Num	Text
43.5	43.5

General Info

Question Use:	Question Level:	Station:
Not Set! Select Initial and/or Continuing!	Not Set! Select 'RO', 'SRO', 'EO', etc!	Not Set! Select 'Braidwood', 'Byron', 'Dresden', etc!

Cognitive Level:	NUREG 1021 Appendix B Information
N/A	Not identified for this question

Associated local objective(s):

	Q25/100 2.4.44

Other NRC Data

References Provided	• EP-AA-1003 Addendum 3 page CL 2-3 • EP-AA-111-F-07 Page 1 with the words "General Emergency" redacted throughout the flowchart.
K/A Justification	This question meets the KA because the examinee has to demonstrate knowledge of emergency plan protective action recommendations to answer the question.
SRO-Only Justification	This question is linked to SRO only task 997777.02 (Given a postulated E-Plan condition, determine and recommend Offsite Protective actions (PARs), IAW corporate EP, and station specific EP procedures). Also linked to 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Additional Information	This question is high cog, written at the analysis/comprehension level. The examinee has to analyze the conditions provided in the stem and then determine when and what actions are required based on that analysis (3-SPK/SPR).
NRC Exams Only	
Question Type	New — **Difficulty** N/A
Technical Reference and Revision #	• EP-AA-111 Rev. 21 • EP-AA-111-F-07 Rev. H • EP-AA-1003 Addendum 3 Rev. 2
Training Objective	LP87537.01.08 Describe Protective Action Recommendations (PARs), including under what conditions they are applicable, and the critical time limits involved
Previous NRC Exam Use	None