EXAMINATION ANSWER KEY

ILT 17-1 NRC SRO Written Exam

1 **ID: 2062725** **Points: 1.00**

A major transient has occurred, all control rods inserted, and plant conditions at 0100 are as follows:·

- Main Steam Line Tunnel temperature is 168°F.
- RPV water level is -100" on the lowest valid RPV level indication and lowering 1" every 5 minutes.
- Suppression pool level indication is pegged low.
- No isolations have been defeated.

At 0130, which of the following methods may be used to depressurize and maintain RPV pressure below 52 psig?

A. Open Main Steam Line Drains.

B. Open the Reactor vessel head vent.

C. Operate Reactor Core Isolation Cooling in tank to tank mode.

D. Use the Reactor Water Cleanup System in the recirculation mode.

Answer: B

Answer Justification / Plausibility Statements
B is correct. IAW CPS 4407.01 Emergency RPV Depressurization, the following Alternate Depressurization Systems are listed in Detail L: • SRVs…open only if suppression pool level is above 8 ft. • Main condenser • TDRFPs • MSL drains • RCIC • RWCU (recirculation mode) (OK to defeat RWCU interlocks) • Head vent • RWCU (reject mode) • Shutdown cooling • SRVs from remote shutdown panel (4003.01)…open only if suppression pool level is above 8 ft. • SRVs from outside the control room (4303.01P004)…open only if suppression pool level is above 8 ft. With the conditions provided in the stem: • MSL Drains have a Group 1 isolation signal due to High Main Steam Line Tunnel temperature will require defeat of isolation signals to be used as an Alternate Depressurization System. • RCIC cannot be used due to high High Main Steam Line Tunnel temperature. • RT isolates at RPV level 2 and will require defeat of isolation signals to be used as an alternate depressurization system. • The reactor vessel head vent is the only listed system that can be used as an alternate depressurization system. Incorrect Responses: A is incorrect but plausible. MSL Drains are listed as an alternate depressurization system but cannot be used due to the Group 1 isolation signal on Main Steam Line Tunnel temperature. C is incorrect but plausible. RCIC is listed as an alternate depressurization system but cannot be used due to the Group 5 isolation signal on Main Steam Line Tunnel temperature. D is incorrect but plausible. RT in recirc mode is listed as an alternate depressurization system but cannot be used due to the RPV water level value provided in the stem.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295030.EA2.01	EA2.01	4.1*	4.2*	1		5

System Name
Low Suppression Pool Water Level

Category Statement
Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : (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

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

Other NRC Data

References Provided:	None		
K/A Justification Statement:	Question meets the KA because the examinee has to demonstrate the ability to interpret the Suppression Pool Level value provided in the stem to determine which depressurization system can be used to depressurize the reactor.		
SRO Only Justification Statement:	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 written at the application and analysis level (high cog). The examinee must evaluate a number of parameters provided in the stem, determine the correct path in EOP-3 and select the correct response. (3-SPK)		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	CPS 4407.01 Rev. 30		
Training Objective:	N-CL-OPS-DB-LP87555.01.09 Given plant conditions after entry to EOP-3 and copies of other EOPs, determine the appropriate actions for these conditions.		
Previous NRC Exam Use:	None		

2	ID: 2029842	Points: 1.00

The plant was operating at rated thermal power.

- At 11:00, Security reports that a group of anti-nuclear protestors have assembled on the intersection of Highway 54 and the CPS access road to demonstrate.
- At 12:00, A situation requiring evacuation of the Main Control Room (MCR) is ordered due to heavy smoke; CPS 4003.01 Remote Shutdown (RS) is entered.
- At 12:05, All MCR actions were taken prior to evacuation and all systems responded as expected; the MCR Operators begin their transit to the Remote Shutdown Panel.
- At 12:16, MCR Operators arrive at the Remote Shutdown Panel, and assume plant control.

What is the highest Emergency Classification for this event?

HU1 Confirmed SECURITY CONDITION or threat.

Emergency Action Level (EAL):

1. Notification of a credible security threat directed at the site as determined per SY-AA-101-132, Security Assessment and Response to Unusual Activities.
   OR
2. A validated notification from the NRC providing information of an aircraft threat.
   OR
3. Notification by the Security Force of a SECURITY CONDITION that does not involve a HOSTILE ACTION.

HU3 FIRE potentially degrading the level of safety of the plant.

Emergency Action Level (EAL):

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Escalation of the emergency classification level would be via IC CA2 or MA5.

1. A FIRE in ANY Table H2 area is not extinguished in < 15-minutes of ANY of the following FIRE detection indications:
   - Report from the field (i.e., visual observation)
   - Receipt of multiple (more than 1) fire alarms or indications
   - Field verification of a single fire alarm

   OR

2. a. Receipt of a single fire alarm in ANY Table H2 area (i.e., no other indications of a FIRE).
   AND
   b. The existence of a FIRE is not verified in < 30 minutes of alarm receipt.

   OR

3. A FIRE within the plant PROTECTED AREA not extinguished in < 60-minutes of the initial report, alarm or indication.

OR

4. A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Table H2 Vital Area
• Containment • Auxiliary Building • Fuel Building • Control Building (excluding Chem Lab) • Diesel Generator & HVAC Building • Screenhouse

HA2 Control Room evacuation resulting in transfer of plant control to alternate locations

Emergency Action Level (EAL):

A Control Room evacuation has resulted in plant control being transferred from the Control Room per CPS 4003.01, Remote Shutdown.

HS2 Inability to control a key safety function from outside the Control Room

Emergency Action Level (EAL):

Note: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

1. A Control Room evacuation has resulted in plant control being transferred from the Control Room per CPS 4003.01, Remote Shutdown.

AND

2. Control of ANY Table H1 key safety function is **not** reestablished in < 15 minutes.

Table H1 Safety Functions
• Reactivity Control (ability to shut down the reactor and keep it shutdown) • RPV water level (ability to cool the core) • RCS Heat Removal (ability to maintain heat sink)

A.	HU1
B.	HU3
C.	HA2
D.	HS2

Answer:	C

Answer Justification / Plausibility Statements
C is correct. Per EP-AA-1003 Addendum 3 Emergency Action Levels for Clinton Station, the Emergency Action Level (EAL) HA2 Control Room evacuation resulting in transfer of plant control to alternate locations is the highest Emergency Classification for this event. CPS 4003.01 Remote Shutdown (RS) was entered at 12:00 to evacuate the Main Control Room (MCR). This meets the requirements for the HA2 declaration. • At time 12:05, the MCR is evacuated. • At time 12:16, Remote Shutdown Panel (RSP) control was established (11 minutes elapsed). Incorrect Responses: A is incorrect but plausible. This response would be correct if the conditions presented in the stem constitutes a confirmed security condition or threat (HU1). A demonstration does not rise to the level of a security condition. Security conditions are defined as a security event that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. B is incorrect but plausible. This response would be correct if the conditions presented in the stem meets the conditions for a fire potentially degrading the level of safety of the plant (HU3). Based on the conditions presented in the stem, the only indication is heavy smoke with no fire alarms; thus entry into HU3 is not required. D is incorrect but plausible. This response would be correct if the conditions presented in the stem constitutes the inability to control a key safety function from outside the control room (HS2). Based on the conditions presented in the stem, control of Table H1 safety functions from the RSP was established within 15 minutes of the last person leaving the MCR; thus entry into HS2 is not required.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295016	B2.4.41	2.9	4.6	1	1	N/A

System Name
Control Room Abandonment

Category Statement
Knowledge of the emergency action level thresholds and classifications. (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):

	Q2/77

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee must demonstrate knowledge of EAL action level thresholds and classifications with respect to a Control Room Abandonment event.		
SRO Only Justification Statement:	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 evaluate the conditions in the stem to determine the correct answer (3-SPR/SPK).		
NRC Exams Only (as applicable)			
Question Type:	Bank (duplicate of CL-LC-0921)	**Difficulty:**	N/A
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: 2029843	Points: 1.00

The plant was operating at Rated Thermal Power (RTP) when a coolant leak into the DW resulted in a reactor scram, entry into EOP-1 RPV Control and EOP-6 Primary Containment Control.

RPV level and pressure are being maintained within the normal control bands of EOP-1 using HPCS.

Then, containment parameters degraded requiring entry into EOP-3 Emergency RPV Depressurization.

Under these conditions:

- LPCS / LPCI injection ____(1)____ required to be terminated and prevented.
- Suppression Pool Level must be verified above a minimum of ____(2)____ before ADS can be initiated.

A.	(1) is (2) 8 ft.

B.	(1) is (2) 15 ft. 1 in.

C.	(1) is NOT (2) 8 ft.

D.	(1) is NOT (2) 15 ft. 1 in.

Answer:	A

Answer Justification / Plausibility Statements
A is correct. Based on the stem: • Entry into EOP-1 and EOP-6 was made due to high Drywell (DW) pressure (above 1.68 psig). • With DW pressure at or above 1.68 psig, per EOP-3 "prevent LPCS and LPCI injection not needed for core cooling". • Since RPV level is currently being maintained within the normal level control band of EOP-1, LPCS and LPCI are not needed for core cooling and therefore should be terminated and prevented. Additionally per EOP-3, ADS may only be initiated if suppression pool level is above 8 ft. Incorrect Responses: B is incorrect but plausible. This response would be correct if a Suppression Pool level of 15 ft. 1 in. was the threshold for initiating ADS. A Suppression Pool level of 15 ft. 1 in. is actually an EOP-6 threshold to enter EOP-3. C is incorrect but plausible. This response would be correct if LPCS and LPCI were needed for core cooling. Per EOP-3 this would preclude the requirement for terminating and preventing LPCS and LPCI injection. D is incorrect but plausible. This response would be correct if: • LPCS and LPCI were needed for core cooling, and • a Suppression Pool level of 15 ft. 1 in. was the threshold for initiating ADS.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295024	B2.1.23	4.3	4.4	1	1	N/A

System Name
High Drywell Pressure

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):

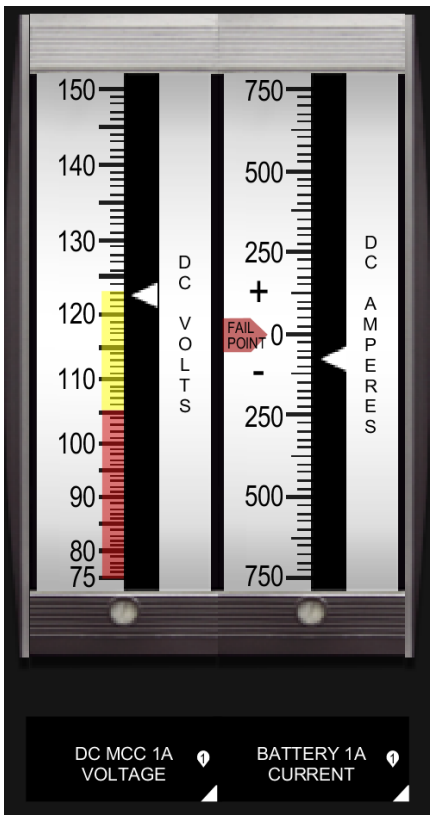
	Q3/78

Other NRC Data

References Provided:	None		
K/A Justification Statement:	Question meets the KA because with regard to High Drywell Pressure, the candidate must demonstrate the ability to perform specific integrated plant procedures (EOP-3) for the conditions presented in the stem.		
SRO Only Justification Statement:	This question is linked to 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency conditions. It is also linked to SRO only task 440701.03 Direct actions to Emergency Depressurize per EOP-3 during non-ATWS.		
Additional Information:	Question is written at the application and analysis level (high cog). The examinee must evaluate a number of parameters provided in the stem, determine the correct path in EOP-3 and select the correct response. (3-SPK)		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 4401.01 (EOP-1), Rev. 30 • CPS 4407.01 (EOP-3), Rev. 30 • CPS 4402.01 (EOP-6), Rev. 30		
Training Objective:	LP87555.01.02 Given a diagram of EOP-3: .02 State when each individual override of EOP-3 is applicable. .03 State the minimum Suppression Pool level necessary for SRV operation.		
Previous NRC Exam Use:	None		

4	ID: 2030692	Points: 1.00

The unit was operating in Mode 2 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.		

For the conditions provided in the stem, which of the below listed 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 Battery Charger 1A has tripped 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). Per ITS 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered, unless a loss of safety function exists. With DC MCC 1A energized as indicated on the meter provided in the stem, a loss of safety function does not exist and ITS 3.8.1 Condition B is not required to be entered. Incorrect Responses: B is incorrect but plausible. This response would be correct if Div 1 Battery was inoperable based on battery current indicating greater than 2 amps, requiring entry into ITS 3.8.4 Condition B. Battery float current is used to determine Battery Charger capacity and operability, not battery operability. C is incorrect but plausible. This response would be correct if the application of ITS LCO 3.0.6 required entry into ITS 3.8.1 Condition B based on the inoperability of the Div 1 DC Battery Charger. D is incorrect but plausible. This response would be correct if: • Div 1 Battery was inoperable based on battery current indicating greater than 2 amps, requiring entry into ITS 3.8.4 Condition B, and • the application of ITS LCO 3.0.6 required entry into ITS 3.8.1 Condition B based on the inoperability of the Div 1 DC Battery Charger. Battery float current is used to determine Battery Charger capacity and operability, not battery operability.

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):

	Q4/79

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to interpret the battery voltage indications provided in the stem to determine required actions in Technical Specifications.		
SRO Only Justification Statement:	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 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 (as applicable)			
Question Type:	Bank (duplicate of CL-ILT-N14077) (previous)	**Difficulty:**	N/A
Technical Reference and Revision #:	• ITS 3.8.4 Amendment 187 • ITS 3.8.1 Amendment 141 • ITS 3.0 Amendment 213 • OP-CL-108-104-1001 Rev. 10d		
Training Objective:	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		
Previous NRC Exam Use:	ILT 14-1 NRC Exam		

5	ID: 2062206	Points: 1.00

An ATWS is in progress.

Plant conditions require lowering RPV water level to reduce reactor power.

At 0200:00, all Detail F1 injection systems were terminated and prevented.

If RPV water level and reactor power trend as shown in the table below, injection will first be re-initiated at _____.

Time	Reactor Power (%)	Wide Range RPV water level (inches)
0200:00	97	+21
0200:10	67	+8
0200:20	40	-10
0200:30	37	-35
0200:40	31	-50
0200:50	26	-60
0201:00	23	-78
0201:10	20	-89
0201:20	18	-101
0201:30	18	-110
0201:40	14	-114
0201:50	13	-132
0202:00	13	-141
0202:10	10	-160 (TAF)
0202:20	7	-160 (TAF)
0202:30	4	-160 (TAF)
0202:40	2	-160 (TAF)

A.	0200:50
B.	0202:00
C.	0202:10
D.	0202:30

Answer:	B

Answer Justification / Plausibility Statements
B is correct. Per 4404.01 ATWS RPV Control, RPV level is lowered to reduce reactor power when all the parameters in the IF AND AND AND THEN override have been exceeded which indicates that excessive energy is being added to the primary containment. RPV level is lowered until the parameters in 4401.01 at 6 have been achieved: • Reactor power below 5%, or • RPV water level below -140", or • All SRV's stay closed and drywell pressure stays below 1.68 psig. These conditions are met at time 0202:00. Incorrect Responses: A is incorrect but plausible. This response would be correct if lowering RPV level to reduce subcooling (Level Band B) which directs injecting at -60" and then controlling RPV level between -60" and TAF. C is incorrect but plausible. This response would be correct if the RPV level criteria for Level Band C (listed above) was TAF instead of -140". Plausible because level is maintained between TAF and -140" in Level Band C. D is incorrect but plausible. This response would be correct if the RPV level criteria for Level Band C (listed above) were TAF and power below 5%.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295037	B2.2.44	4.2	4.4	1	1	N/A

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

Category Statement
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)

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

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

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee must demonstrate the ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions (RPV water level) during an ATWS.		
SRO Only Justification Statement:	This question is linked to 10CFR55.73(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency conditions.		
Additional Information:	This is a high cog question written at the analysis and application level. The examinee has to evaluate the conditions in the stem to determine the correct answer (3-SPR/SPK).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 4404.01 (EOP-1A), Rev. 30		
Training Objective:	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		

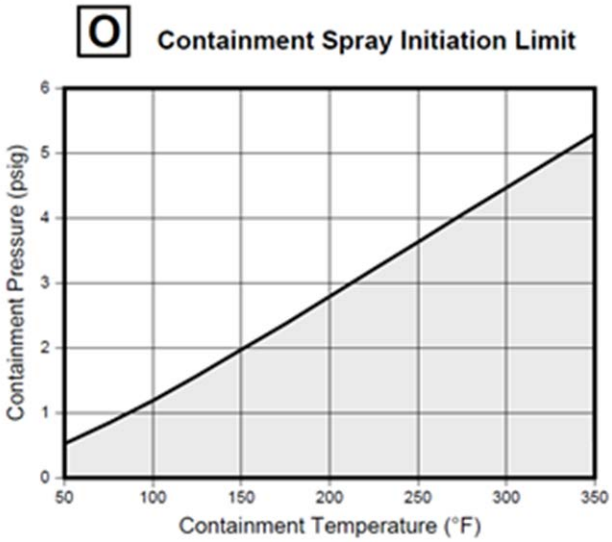
6	ID: 2029851	Points: 1.00

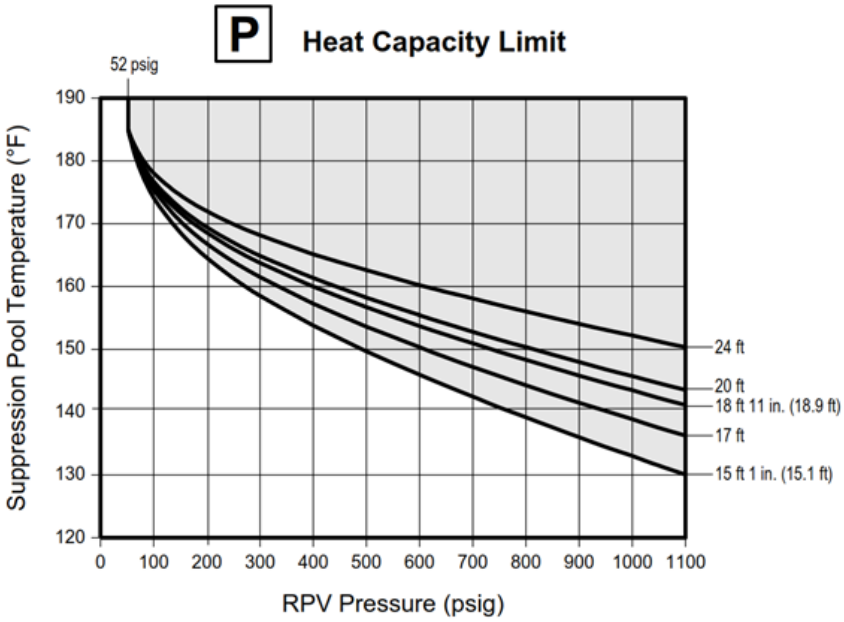
The reactor was at rated power when an event occurred.

Current plant conditions are as follows:

Parameter	Value	Trend
Reactor Power	4%	Steady
Suppression Pool Temperature	145°F	Rising
RPV Pressure	930 psig	Cycling on SRVs
Suppression Pool Level	19'3"	Steady
Containment Temperature	125°F	Rising
Containment Pressure	2.3 psig	Rising
RPV Level	35 inches	Steady

All appropriate EOPs have been entered for the current conditions.





Which action(s) is(are) required by the EOPs?

A.	Perform a blowdown ONLY.
B.	Perform a blowdown AND initiate containment spray.
C.	Lower and maintain RPV Pressure 500 - 600 psig ONLY.
D.	Lower and maintain RPV pressure 500 - 600 psig AND initiate containment spray.

Answer:	D

Answer Justification / Plausibility Statements
D is correct: Per conditions presented in the stem and Figure O Containment Spray Initiation Limit, you are in the "OK to spray" region and therefore should initiate containment sprays. AND Per EOP-6 Primary Containment Control, IF suppression pool temperature cannot be held below Fig P, Heat Capacity Limit AND you are NOT in RPV flooding, THEN lower RPV pressure to stay below the limit of Fig P. With Suppression Pool level at 19'3", the 18' 11" curve of Figure P Heat Capacity Limit is used. At 930 psig RPV pressure, the heat capacity limit is ~ 145°F. Lowering RPV pressure 500-600 psig would raise the heat capacity limit to 154°F - 157°F. Incorrect Responses: A is incorrect but plausible. This response would be correct if: • Figure O Containment Spray Initiation Limit prohibited initiating containment sprays, and • EOP-6 Primary Containment Control prohibited lowering RPV pressure to stay below the limit of Figure P Heat Capacity Limit based on conditions presented in the stem. B is incorrect but plausible. This response would be correct if EOP-6 Primary Containment Control prohibited lowering RPV pressure to stay below the limit of Figure P Heat Capacity Limit based on conditions presented in the stem. C are incorrect but plausible. This response would be correct if Figure O Containment Spray Initiation Limit prohibited initiating containment spray based on conditions presented in the stem.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295026.EA2.01	EA2.01	4.1*	4.2*	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 water 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):

	Q6/81

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee must demonstrate the ability to interpret suppression pool water temperature as it applies to SUPPRESSION POOL HIGH WATER TEMPERATURE in order to select the correct response.		
SRO Only Justification Statement:	The question is SRO ONLY level because the evaluation of plant parameters (high suppression pool temperature) require the use of details on EOP-6. The interpretation of the data and applying it to the limits of these details is an SRO only function at Clinton Power Station. Also linked to SRO only task 440201.10 (Respond to a high Suppression Pool temperature per EOP-6). Also related to 10CFR55.43(b)(5) for assessment of facility conditions.		
Additional Information:	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the parameters in the table and Figures O & P, and then determine how the plant is impacted by that analysis to answer the question (3-SPK/SPR).		
NRC Exams Only (as applicable)			
Question Type:	Modified (From CL-ILT-N14078)	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 4402.01 (EOP-6), Rev. 30		
Training Objective:	LP87558.01.19 Given plant conditions after entry to EOP-6 and copies of other EOPs, determine the appropriate actions for these conditions. 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		

7	ID: 2057861	Points: 1.00

The plant was operating at rated thermal power when two trouble conditions were received on 1H13-P841 FP XL3 Panel:

- 84-01 FB 712 HPCS Rm Col 102-108 AE-AF INPUT DEVICE NO RESPONSE
- 84-02 FB 712 HPCS Rm Col 102-108 AE-AF INPUT DEVICE NO RESPONSE

A reset attempt of XL-3 was performed, but the trouble conditions would NOT reset.

The required action is to...

A.	Enter and execute CPS 1893.04 Fire Fighting.
B.	Restore BOTH detectors to a FUNCTIONAL status within 14 days.
C.	Establish a fire watch to patrol the HPCS Pump Room once per hour.
D.	Restore AT LEAST ONE detector to a FUNCTIONAL status within 14 days.

Answer:	C

Answer Justification / Plausibility Statements

C is correct:

Per CPS 3213.02M001 XL3 System Matrix, page 33 lists devices 84-01, 84-02, and 84-03 as providing detection for the HPCS Pump Room.

Per CPS 3213.02 Plant Fire Detection System, step 2.2.1.10, a TROUBLE condition is defined as a signal indicating trouble of any nature, such as a circuit break or ground, occurring in the devices or wiring associated with a local Protection/Detection system. An alarm (step 2.2.1.9) is a signal indicating an emergency requiring immediate action. Alarms are required to be investigated, and if an actual fire exists, mitigated using CPS 1893.04 Fire Fighting.

Per CPS 3213.02 Appendix C page 37, INPUT DEVICE NO RESPONSE is a trouble message indicating that no supervisory data is being received from the detectors listed in the stem.

Per CPS 1893.01 Fire Protection Impairment Reporting, step 8.1.1 states that whenever a fire protection function is deemed NONFUNCTIONAL, or thought to be NON-FUNCTIONAL, Appendix A shall be consulted for the required interim measures to compensate for the NON-FUNCTIONAL system.

CPS 1893.01 Fire Protection Impairment Reporting, Appendix A section 9 lists the FUNCTIONALITY requirements for the fire detection instrumentation listed in Appendix D. Since HPCS is required to be OPERABLE in Mode 1, the fire detection instrumentation for the HPCS room is required to be FUNCTIONAL. Appendix D PAGE 38 lists 3 Function A fire detection instruments (smoke detectors) in Fuel Building Zone F-1b. With 2 of the 3 detectors NON-FUNCTIONAL, measure b. applies (more than 1/2 of the Function A fire detection instruments) which requires the establishment of a fire watch to patrol the area once per hour.

Incorrect Responses:

A is incorrect but plausible. This answer would be correct if fire alarms (not troubles) were received and an actual fire existed.

B is incorrect but plausible. This answer would be correct if the number of non-functional detectors constituted no more than half of the total in the fire zone (1893.01 Appendix A Section 9 Measure A. With the requirement to establish a fire watch, there are no requirements to restore detector functionality within 14 days.

D is incorrect but plausible. This answer would be correct if 1893.01 permitted reverting to Measure A (less than half of the detectors non-functional) if a single detector in the stem was restored to a functional status. Incorrect because the answer excludes establishing a fire watch patrol, which is required. With the requirement to establish a fire watch, there are no requirements to restore detector functionality within 14 days.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
600000.AA2.09	AA2.09	2.4	2.8	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
That a failed fire alarm detector exists

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):

	Q7/82

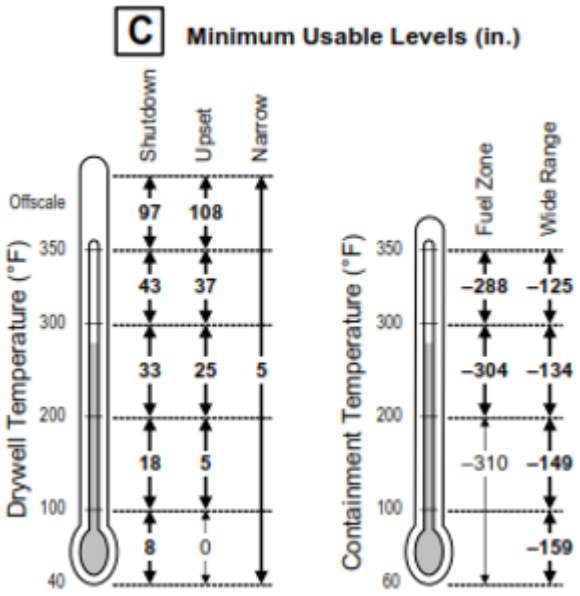
Other NRC Data

References Provided:	CPS 1893.01 in it's entirety with discussion of FM-200 on page 26 redacted to eliminate providing cues with Q49 on the RO exam.		
K/A Justification Statement:	This question meets the KA because the examinee has to demonstrate the ability to determine and interpret a failed fire alarm detector exists as it applies to PLANT FIRE ON SITE to answer the question.		
SRO Only Justification Statement:	This question is linked to SRO only task 189301.01 Perform Fire Protection Impairment Evaluations.		
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 required actions using a reference (3-SPR).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 3213.02M001 Rev. 2c • CPS 3213.02 Rev. 5d • CPS 1893.01, Rev. 20f • CPS 1893.04, Rev. 18c		
Training Objective:	LP85804.2.4.25 Knowledge of fire protection procedures.		
Previous NRC Exam Use:	None		

8	ID: 2029863	Points: 1.00

The plant was operating at rated thermal power when a transient occurred resulting in a scram. Conditions are as follows:

- RPV Pressure is 850 psig and lowering quickly
- Containment temperature is 122°F and rising slowly
- RPV level is lowering slowly



If RPV level reaches _____(1)_____, a blowdown will be required _____(2)_____.

A.	(1) -149 inches on 1B21-R623A Rx Level & Press Recorder A (2) immediately

B.	(1) -149 inches on 1B21-R623A Rx Level & Press Recorder A (2) within the next 10 minutes to determine if alternate injection systems can restore level

C.	(1) -162 inches on B21-R615 Rx Fuel Zone Level (2) immediately

D.	(1) -162 inches on B21-R615 Rx Fuel Zone Level (2) within the next 10 minutes to determine if alternate injection systems can restore level

Answer:	A

Answer Justification / Plausibility Statements
A is correct. Per CPS 4401.01 RPV Control (EOP-1) Detail C Minimum Usable Levels (in.), with containment temperature > 100°F the minimum usable level for the Wide Range RPV level instruments is -149 inches. Additionally, EOP-1 requires an immediate blowdown when RPV level reaches TAF. Incorrect Responses: B is incorrect but plausible. This response would be correct if the 10 minute 'stay' period for evaluating plant response (CPS 4411.07 RPV Level Instrumentation) was allowed based on conditions presented in the stem. Per CPS 4411.07 step 4.4.1, a blowdown shall be initiated when RPV water level reaches TAF. There is no allowance for momentary level excursions below TAF when in EOP-1 level control, therefore, the 10 minute stay period does not apply to level at or below TAF. C is incorrect but plausible. This response would be correct if the Fuel Zone indication was reliable based on conditions presented in the stem. Per EOP-TB, the Fuel Zone instruments are calibrated for an RPV pressure of 0 psig with no jet pump flow. Any flow through the jet pumps, including that caused by SRV actuation, will result in erratic Fuel Zone indication. D is incorrect but plausible. This response would be correct if: • the Fuel Zone indication was reliable, and • the 10 minute 'stay' period for evaluating plant response (CPS 4411.07 RPV Level Instrumentation) was allowed based on conditions presented in the stem. Per CPS 4411.07 step 4.4.1, a blowdown shall be initiated when RPV water level reaches TAF. There is no allowance for momentary level excursions below TAF when in EOP-1 level control, Therefore, the 10 minute stay period does not apply to level at or below TAF.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295011	B2.4.03	3.7	3.9	1	2	N/A

System Name
High Containment Temperature (Mark III Containment Only)

Category Statement
Ability to identify post-accident instrumentation. (CFR: 41.6 / 45.4)

K/A Statement
N/A

CFR Data

10CFR55-41b (RO) Data

Para Num	Text
41.6	41.6

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):

	Q8/83

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to demonstrate the ability to identify the correct post-accident instrumentation used to make operational decisions with a high containment temperature in order to determine the correct response.		
SRO Only Justification Statement:	This question requires the examinee to assess plant conditions and to know the content of procedures in order to select a required course of action. Linked to 10CFR55.43(b)(5) and SRO only task 100509.16D (Demonstrate knowledge of symptom based EOP mitigation strategies).		
Additional Information:	Question is written at the application and analysis level (high cog). The examinee must analyze conditions presented in the stem and using references determine when a blowdown is required. (3-SPK/SPR)		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 4401.01 Rev. 30 • CPS 4411.07 Rev. 6b • EOP-TB Rev. 7		
Training Objective:	LP87552.01.07 Given specified plant conditions and a diagram of EOP-1, be able to properly implement the following inserts per EOP-1 without error: .05 Figure C, Minimum Usable Levels LP87552.01.08 Given specified plant conditions and a diagram of EOP-1, be able to properly prioritize and implement the following actions for EOP-1 without error: .04 Deciding when you may anticipate Blowdown, and when you should perform Blowdown		
Previous NRC Exam Use:	None		

9	ID: 2033265	Points: 1.00

The plant was operating at rated thermal power when the following annunciators were received:

- 5067-8L SRV MONITORING SYSTEM TROUBLE
- 5066-5B ADS OR SAFETY RELIEF VALVE LEAKING
- 5004-3F SPDS CSF ALARM (Suppression Pool Temperature alarming 2°F above the setpoint)

Under these conditions, the CRS shall direct placing _____(1)_____ Residual Heat Removal loop(s) into suppression pool cooling.

If Suppression Pool temperature rises an additional 3°F over the next 24 hours, ITS actions require entry into CPS _____(2)_____.

A.	(1) one (2) 3006.01 Unit Shutdown

B.	(1) one (2) 4100.01 Reactor Scram

C.	(1) two (2) 3006.01 Unit Shutdown

D.	(1) two (2) 4100.01 Reactor Scram

Answer:	C

Answer Justification / Plausibility Statements
C is correct. Per CPS 4402.01 Primary Containment Control (EOP-6) a suppression pool temperature above 95°F is an EOP-6 entry and will cause the PRI-CNMT CSF box to turn red. If you cannot hold suppression pool temperature below 95°F, THEN start all available suppression pool cooling. Based an conditions presented in the stem, neither RHR pump is needed for core cooling or containment spray and therefore both are available for suppression pool cooling. Per ITS 3.6.2.1 Suppression Pool Average Temperature Required Action B.1, if suppression pool average temperature cannot be restored to ≤ 95°F within 24 hours then THERMAL POWER must be reduced ≤ 1% RTP within 12 hours requiring entry into CPS 3006.01 Unit Shutdown. Incorrect Responses: A is incorrect but plausible. This answer is partially correct in that the plant must be shutdown, however EOP-6 requires RHR to be placed in suppression pool cooling but requires starting all available suppression pool cooling, which would include starting both RHR 'A' and 'B' subsystems. B is incorrect but plausible: • Part 1 - EOP-6 requires RHR to be placed in suppression pool cooling but requires starting all available suppression pool cooling, which would include starting both RHR 'A' and 'B' subsystems. • Part 2 - ITS requires the plant to be scrammed if suppression pool average temperature is > 110°F but ≤ 120°F (ITS 3.6.2.1 D.1). This answer is incorrect because SP temperature has not yet reached this value (currently at 100°F). D is incorrect but plausible. ITS requires the plant to be scrammed if suppression pool average temperature is > 110°F but ≤ 120°F (ITS 3.6.2.1 D.1). This answer is incorrect because SP temperature has not yet reached this value (currently at 100°F).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
295013.AA2.01	AA2.01	3.8	4.0	1		5

System Name
High Suppression Pool Temperature

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

K/A Statement
Suppression pool 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):

	Q9/84

Other NRC Data

References Provided:	None		
K/A Justification Statement:	Question meets the KA because the examinee must demonstrate the ability to determine and/or interpret Suppression Pool Temperature as it applies to HIGH SUPPRESSION POOL TEMPERATURE in order to select the correct response.		
SRO Only Justification Statement:	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:	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the conditions provided in the stem and then determine which actions based on EOPs and ITS (3-SPK).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 4402.01 (EOP-6) Rev. 30 • ITS 3.6.2.1 (3.6-28/29) Amend. 95		
Training Objective:	700003.06 Given an PPC system Annunciator, DESCRIBE: a. The condition causing the annunciator b. Any automatic actions c. Any operational implications		
Previous NRC Exam Use:	None		

EXAMINATION ANSWER KEY

ILT 17-1 NRC SRO Written Exam

10	ID: 2057867	Points: 1.00

The plant was operating at rated thermal power when annunciator 5013-5D HIGH-HIGH LEVEL FLR/EQUIP DRN SUMP - AUX BLDG was received.

The 'A' RO reports:

- computer point CM-BC813 RCIC Floor Drn Sump Lvl indicates HI HI, and
- Suppression Pool level is lowering at 1 inch per minute.

CPS 4304.01 Flooding Table 2 - Suppression Pool Level / ECCS Room Equalization Levels

ECCS Room	Final Pool Level		ECCS Room	Final Pool Level
HPCS	~ 15' 3"		RHR A	~ 15' 5"
LPCS	~ 14' 4"		RHR B	~ 15' 5"
RCIC	~ 16' 2"		RHR C	~ 14' 10"

If Suppression Pool level lowers to the equalization level, what EOP action is required and what is the reason for that action?

A.	Scram to reduce the break flow into the secondary containment.

B.	Enter EOP-3 and blowdown to prevent exceeding the heat capacity of the suppression pool.

C.	Shutdown the reactor due to the wide spread and immediate threat to equipment in the secondary containment.

D.	Isolate the discharge into the RCIC Pump Room to protect equipment necessary for safe operation of the plant.

Answer:	D

Answer Justification / Plausibility Statements

D is correct:

Per CPS 4406.01 Secondary Containment Control, an EOP-8 entry condition has been exceeded due to floor drain sump level above max normal (Table V). If floor drain sump level cannot be held below max normal, then the required action is to isolate all discharges into affected areas except systems needed for damage control or EOP actions. Per the EOP-TB page 9-1, EOP-8 Secondary Containment Control is used to protect equipment in the secondary containment.

Additional actions in 4406.01 Secondary Containment Control are not required because:

- The leak from the Suppression Pool does not constitute a primary leak, precluding the need to perform actions at 32, and
- multiple areas of the secondary containment are not impacted by the leakage (only the RCIC room) requiring the reactor to be shutdown.

EOP-6 Primary Containment Control will also be entered due to Suppression Pool Level below 18.9 feet. Actions are required to initiate normal Suppression Pool makeup methods to hold level above 15.1 ft and to scram and enter EOP-3 blowdown if SP level cannot be maintained above 15.1 ft. Since the equalization level for the RCIC Room is 16' 2", a scram and blowdown will not be required.

Incorrect Responses:

A is incorrect but plausible. Per EOP-TB page 9-11 and 9-12, break flow into the secondary containment is reduced when a primary system is discharging into the secondary containment by inserting a scram. Since the leak in the stem is not a primary system, inserting a scram to reduce the break flow is not the action or the reason for performing required EOP actions.

B is incorrect but plausible. EOP-6 requires a blowdown to be performed if Suppression Pool level cannot be held above 15'1" to prevent exceeding the heat capacity of the Suppression Pool. Since the equalization level is above 15'1", performing a blowdown to prevent exceeding the heat capacity limit of the suppression pool is not the action or the reason for performing required EOP actions.

C is incorrect but plausible. This answer would be correct if the SP leakage was impacting another area in the secondary containment (cross flooding). EOP-8 requires a reactor shutdown to be performed if 2 or more areas in the secondary containment are above max safe values for level due to non-primary system leakage. EOP-TB page 9-10 states the bases for this action is due to the indications of a wide-spread problem posing a direct and immediate threat to the secondary containment. Since only 1 area is impacted (the RCIC Pump Room), a reactor shutdown is not required by EOP-8.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.295036	B2.1.32	3.8	4.0	1	2	N/A

System Name
Secondary Containment High Sump/Area Water Level

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):

	Q10/85

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee must demonstrate the ability to explain and apply system limits and precautions with regard to Secondary Containment High Sump/Area Water Level to answer the question correctly.		
SRO Only Justification Statement:	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:	This is a high cog question written at the analysis and comprehension level. The examinee has to analyze the conditions provided in the stem and then determine the effect/consequences/mitigation strategies based on that analysis (3-SPK).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 4304.01 Rev. 6c • CPS 4402.01 (EOP-6) Rev. 30 • CPS 4406.01 (EOP-8) Rev. 30 • EOP-TB Rev. 007		
Training Objective:	N-CL-OPS-DB-LP87559.01.10 Describe the normal means of control for a slow increase in Secondary Containment Water Levels.		
Previous NRC Exam Use:	None		

11	ID: 2057277	Points: 1.00

The plant is operating at 100% rated thermal power.

Which of the following scenarios would require entry into ITS 3.3.1.1 RPS Instrumentation Condition A (one or more Functions with one channel inoperable)?

Scenario	Event
1	5 of the 8 LPRMs on APRM 'A' Axial Level A are failed downscale.
2	It is discovered that Reactor Recirculation Pump 'A' breakers 3A and 4A are incapable of opening.
3	During performance of an RPS Channel Functional Surveillance, one of the instrument channels tripped outside the allowable value listed in ITS Table 3.3.1.1-1 Reactor Protection System Instrumentation.
4	During performance of Scram Discharge Volume Vent/Drain valve stroke time testing, 1C11-F010 Scram Discharge Volume Vent Valve closed, but exceeded the section 9.1 Acceptance Criteria time limit listed in the surveillance.

Scenario...

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

Answer:	C

Answer Justification / Plausibility Statements

C is correct.

Per ITS B3.3.1.1 RPS Instrumentation, The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have four OPERABLE channels, **with their setpoints within the specified Allowable Value**, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Incorrect Responses

A is incorrect but plausible due to the large number of LPRMs that are failed. Per ITS B 3.3.1.1 Functions 2.a and 2.b, 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 and Average Power Range Monitor Flow Biased Simulated, Thermal Power–High 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.** The stem states that 5 of the 8 LPRMs on APRM 'A' Axial Level 'A' are failed, which leaves 3 operable, which is greater than the number of LPRMs required to consider APRM 'A' OPERABLE. This distracter would be correct if 2 additional LPRMs were failed.

B is incorrect but plausible. RR Pump 'A' 3A and 4A breakers will trip if an EOC/RPT signal is received. The EOC/RPT signal is generated from a Turbine Stop Valve or Turbine Control Valve closure signal, which will generate an RPS trip signal (ITS 3.3.1.1 RPS Instrumentation Functions 9 and 10). The trip of the RR Pumps, however, is not an RPS function, so failure of the RR 3A and 4A breakers would not require entry into ITS 3.3.1.1. The failure would require entry into ITS 3.3.4.1 EOC-RPT Instrumentation.

D is incorrect but plausible. The Scram Discharge Volume Vent and Drain Valves automatically close on an RPS initiation and are stroke time tested per CPS 9031.15 Reactor Mode Switch Functional Test. Per CPS 9031.15, the operability requirements for the SDV V&D Valves are tied to ITS SR 3.1.8.3 Scram Discharge Volume Vent and Drain Valves and is based on limiting the amount of reactor coolant discharged during a scram. The valves are required to close to satisfy the requirements of ITS 3.3.1.1, but the stroke time requirement only applies to ITS 3.1.8.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.212000	B2.2.25	3.2	4.2	2	1	N/A

System Name
Reactor Protection 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):

	Q11/86

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee must demonstrate knowledge of the ITS bases for the RPS system to answer the question correctly.		
SRO Only Justification Statement:	SRO justification - this question is linked to SRO only task 140109.23 Apply The Administrative Requirements For Execution Of Technical Specifications Or Off-Site Dose Calculation Manual Requirements and 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases. This question also requires knowledge of TS bases to analyze TS required actions and terminology.		
Additional Information:	This is a high cog question written at the analysis and application level. The examinee has to recall several pieces of information and then compare that with knowledge of the bases for RPS instrumentation technical specifications to answer the question (3-SPK).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 9031.15 Rev. 27b • ITS B3.3.1.1 (page B3.3-3) Revision No. 10-6 • ITS B3.3.1.1 (page B3.3-7) Revision No. 7-5 • ITS B3.3.1.1 page B3.3-8) Revision No. 0		
Training Objective:	212000.14 Given Reactor Protection System (RPS) and Alternate Rod Insertion (ARI) System operability status and a copy of Tech Specs, DISCUSS the bases for the Reactor Protection System (RPS) and Alternate Rod Insertion (ARI) System Tech Spec LCO, related safety limits and Limiting Safety System Settings.		
Previous NRC Exam Use:	None		

12	ID: 2062265	Points: 1.00

Today is 3/5/18. NO LCO's are in effect.

The plant is operating at rated thermal power with IMD scheduled to perform CPS 9532.08C MSL Tunnel Temp. 1E31-N604C Channel Functional on your shift.

Events occurred as follows:

Time	Event
0830	IMD commences 9532.08C on instrument 1E31-N604C MSL Tunnel Temp Channel C. 6 hour delay clock for entry into associated Conditions and Required Actions of ITS 3.3.6.1 commenced for performance of required surveillance.
0845	IMD reports that 1E31-N604C will NOT trip. CRS directs IMD to suspend performance of 9532.08C until troubleshooting can be performed.

Which of the following cases describes:

1) actions required to be taken by Technical Specifications, and

2) whether the associated function maintains isolation capability?

Case	Required ITS action	Is isolation capability maintained?
1	Perform ITS 3.3.6.1 Required Action A.1 by 3/7/18 0845	yes
2	Perform ITS 3.3.6.1 Required Action A.1 by 3/7/18 0845	no
3	Perform ITS 3.3.6.1 Required Action A.1 by 3/7/18 1430	yes
4	Perform ITS 3.3.6.1 Required Action A.1 by 3/7/18 1430	no

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

Answer:	A

Answer Justification / Plausibility Statements

A is correct.

Per ITS B 3.3.6.1 Primary Containment and Drywell Isolation Instrumentation, page 3.3-142 and 143, Main Steam Tunnel Ambient Temperature - High and Main Steam Line Turbine Building Temperature - High, ambient temperature signals are initiated from thermocouples located in the area being monitored. Four channels of Main Steam Tunnel Temperature–High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each Function has one temperature element.

Per Note 2 of ITS 3.3.6.1 (page 3.3-54), when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains isolation capability.

Since MSL Isolation capability is maintained by the remaining MSL Ambient Temperature channels, the 6 hour surveillance clock can be used. However, once IMD reports that 1E31-N604C will not trip and the surveillance is suspended, the 6 hour clock to delay actions is stopped and Required Actions of ITS 3.3.6.1 must be taken.

Therefore, ITS 3.3.6.1 Action A.1 must be entered immediately (at 3/5/18 0845), and must be performed within 48 hours of the channel being declared inoperable (3/7/18 at 0845). For part 2, with a single instrument failure, isolation capability is maintained for the Main Steam Tunnel Temperature-High Function as stated above.

Incorrect Responses:

B is incorrect but plausible. This answer would be correct if the failure of 1E31-N604C resulted in a loss of isolation capability.

C is incorrect but plausible. This answer would be correct if the 6 hour surveillance clock was allowed to continue past the discovery of inoperability time (3/5/18 0845) to determine when ITS 3.3.6.1 A.1 must be completed. 3/5/18 0830 + 6 hours + 48 hours = 3/7/18 1430.

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

- the 6 hour surveillance clock was allowed to continue past the discovery of inoperability time (3/5/18 0845) to determine when ITS 3.3.6.1 D.1 must be completed (3/5/18 0830 + 6 hours + 48 hours = 3/7/18 1430), and
- the failure of 1E31-N604C resulted in a loss of isolation capability.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
223002.A2.06	A2.06	3.0	3.2	2		5

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

Category Statement
Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; 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
Containment instrumentation failures

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):

	Q12/87

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to predict the impact of a Primary Containment Isolation System instrument failure and then determine which ITS Required Actions apply to answer the question.		
SRO Only Justification Statement:	This question is linked to SRO only task 140109.23 Apply The Administrative Requirements For Execution Of Technical Specifications Or Off-Site Dose Calculation Manual Requirements and 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.		
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 and determine applicability of required actions based on that analysis (3-SPK/SPR).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• ITS 3.3.6.1 (3.3-48 and 3.3-49) Amendment No. 95 • ITS 3.3.6.1 (3.3-54) Amendment No. 192 • ITS B3.3.6.1 (B 3.3-142) Revision No. 0 • ITS B3.3.6.1 (B 3.3-143) Revision No. 1-1 • CPS 9532.08C Rev. 1a		
Training Objective:	223002.12 Given CRVICS 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		

13	ID: 2030630	Points: 1.00

Plant conditions are as follows:

- Mode 2
- Reactor heatup is in progress

(1) In which of the following scenarios is ITS 3.3.1.1 Reactor Protection System (RPS) Instrumentation required to be entered?

Scenario	IRM 'A'	IRM 'B'	IRM 'C'	IRM 'D'	IRM 'E'	IRM 'F'	IRM 'G'	IRM 'H'
A	Range 1 4/125	Range 7 30/125	Range 7 25/125	Range 7 30/125	Range 1 4/125	Range 7 40/125	Range 6 60/125	Range 7 35/125
B	Range 1 4/125	Range 1 4/125	Range 7 30/125	Range 6 60/125	Range 7 30/125	Range 7 35/125	Range 6 60/125	Range 7 30/125

(2) Will ITS 3.0.4 allow the startup to continue into Mode 1 if ITS 3.3.1.1 actions are taken?

A.	(1) A (2) Yes

B.	(1) A (2) No

C.	(1) B (2) Yes

D.	(1) B (2) No

Answer:	A

Answer Justification / Plausibility Statements

A is correct:

Per CPS 5005.03 (3J) IRM DNSC, the IRM Downscale Trip setpoint is 5/125 scale. In scenario A, IRM downscale trips exist on IRM 'A' and IRM 'E'. In scenario B, IRM downscale trips exist on IRM 'A' and IRM 'B'.

The IRM channels are arranged as follows:

- Div 1 - IRM 'A' and 'E'
- Div 2 - IRM 'B' and 'F'
- Div 3 - IRM 'C' and 'G'
- Div 4 - IRM 'D' and 'H'

Per ITS B 3.3.1.1 RPS Instrumentation, the IRM System is divided into four groups of IRM channels, with two IRM channels inputting to each trip logic division. **Per the analysis of Reference 6, six IRM channels, including at least one IRM channel per trip logic division, are required for IRM OPERABILITY.**

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.

Since ITS 3.3.1.1 Action A.1 requires the inop channel to be placed in trip within 48 hours and then allows continued operation for an unlimited period of time, entry into Mode 1 is permitted.

Incorrect Responses:

B is incorrect but plausible if the 48 hour completion time for ITS 3.3.1.1 A.1 is applied to ITS 3.0.4.a.

C is incorrect but plausible since ITS 3.3.1.1 contains the requirement for at least one IRM Channel per trip logic division for IRM operability. Incorrect because IRMs 'A' and 'B' are in different trip logic divisions.

D is incorrect but plausible:

- since ITS 3.3.1.1 contains the requirement for at least one IRM Channel per trip logic division for IRM operability. Incorrect because IRMs 'A' and 'B' are in different trip logic divisions.
- if the 48 hour completion time for ITS 3.3.1.1 A.1 is applied to ITS 3.0.4.a.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
215003.A2.04	A2.04	3.7	3.8	2		7

System Name
Intermediate Range Monitor (IRM) System

Category Statement
Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) 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
Up scale or down scale 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):

	Q13/88

Other NRC Data

References Provided:	• ITS 3.0 (pages 3.0-1 and 3.0-2) • ITS 3.3.1.1 (page 3.3-1)		
K/A Justification Statement:	This question meets the KA because the examinee is required to predict the impact of IRM downscale trips on the RPS system, and then determine whether ITS 3.0.4 allows entry to Mode 1.		
SRO Only Justification Statement:	This question requires the examinee to assess plant conditions and apply the requirements of Tech Specs. Linked to task SRO only task 999999.07 Apply Technical Specification Requirements and 10CFR55.43(b)(2) Facility operating limitations in the Technical Specifications and their bases.		
Additional Information:	This is a high cog question written at the application and comprehension level. The examinee has to determine the system impact based on consequences of the conditions provided in the stem and then determine appropriate actions to take (3-SPK/SPR).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 5005.03 (3J) Rev. 31c • ITS 3.3.1.1 (3.3-1) Amendment No. 95 • ITS B3.3.1.1 (B 3.3-5) Rev. No. 10-8 • ITS 3.0 (3.0-1) Amendment No. 163		
Training Objective:	215003.14 Given IRM System operability status and a copy of Tech Specs, DISCUSS the bases for the IRM System Tech Spec LCO, related safety limits and Limiting Safety System Settings.		
Previous NRC Exam Use:	None		

14	ID: 2049257	Points: 1.00

The plant was operating at rated thermal power when CB MCC A faulted, causing a loss of the AC power source to NSPS Solenoid (RPS) Inverter 'A'.

The NSPS Solenoid (RPS) Inverter 'A' ____(1)____.

To return the 'A' NSPS Solenoid (RPS) bus back to a normal alignment, you would direct the restoration of the lost power source ____(2)____.

A.	(1) is OPERABLE (2) ONLY

B.	(1) is INOPERABLE (2) ONLY

C.	(1) is OPERABLE (2) AND the manual transfer of RPS Distribution Panel Loads (NSPS Solenoid) (RPS) FROM Alternate Power TO the Inverter

D.	(1) is INOPERABLE. (2) AND the manual transfer of RPS Distribution Panel Loads (NSPS Solenoid) (RPS) FROM Alternate Power TO the Inverter

Answer:	A

Answer Justification / Plausibility Statements
A is correct. • ITS 3.8.7 Inverters - Operating covers the operability requirements for the two NSPS Solenoid (RPS) Inverters and the four divisional NSPS inverters. • Per ITS B3.8.7 and CPS 9082.02 Electrical Distribution Verification, the two NSPS Solenoid (RPS) 120V AC Solenoid Buses are OPERABLE if inverter voltage, frequency, and alignment is verified correct to the solenoid bus. If the normal AC power source to the inverter is lost, the DC source to the inverter would supply power to the inverter via an isolation diode to maintain the inverter energized. • Per CPS 3509.01 section 8.5.2 NSPS Solenoid Bus (RPS) Operation With a Single Power Supply, restoration of the lost power source is the only action required to restore the bus to a normal alignment. Incorrect Responses: B is incorrect but plausible. This response would be correct if the NSPS Solenoid (RPS) inverter was rendered INOPERABLE by a loss of the AC power source (like the divisional NSPS inverters). Per ITS B3.8.7, for a NSPS inverter to be considered OPERABLE, the inverter must be powered from the required Class 1E DC bus. The RPS inverter is only required to be powered (by either source) such that its output is within the design voltage and frequency tolerances. C is incorrect but plausible. This response would be correct if the NSPS Solenoid (RPS) inverter had to be manually transferred to restore it to a normal alignment once power was restored (like the NSPS inverters). Since the RPS inverters are powered from both AC and DC sources, the loss of the AC source does not result in the loss of inverter input power, and thus does not require manual transfer of distribution panel loads back to the AC source. D is incorrect but plausible. This response would be correct if: • the RPS inverter was rendered INOPERABLE by a loss of the AC power source, and • the RPS inverter had to be manually transferred to restore it to a normal alignment once power was restored.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
262002.A2.01	A2.01	2.6	2.8	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
Under 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):

	Q14/89

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because it requires the examinee to predict the impact of a loss of the AC power source to the RPS Inverter and based on that prediction, determine which procedure is used to correct, control, or mitigate the undervoltage condition.		
SRO Only Justification Statement:	This question requires knowledge of the ITS bases to determine the operability status of the RPS Inverter. The second part of the question can be answered using RO knowledge of how the RPS Inverter is impacted by a loss of the AC power source to the inverter. 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:	This is a high cog question written at the application and comprehension level. The examinee has to determine operability status based on consequences of the conditions provided in the stem and then determine the required actions necessary to restore the inverter back to a normal lineup (3-SPK).		
NRC Exams Only (as applicable)			
Question Type:	Bank (Duplicate CL-ILT-A12086)	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 3509.01 Rev. 21 • CPS 9082.02 Rev. 36a • ITS B3.8.7 Rev. 5-4		
Training Objective:	700004.03 DESCRIBE the function, operation, interlocks, trips, physical location, and power supplies of the following NSPS System components. .3 NSPS Solenoid Bus Inverter		
Previous NRC Exam Use:	None		

15	ID: 2078969	Points: 1.00

The plant was operating at rated thermal power when a loss of DC MCC 1A occurred.

Which one of the following identifies the effect this will have on the RCIC system and what action is required to be taken per Tech Spec 3.5.3 RCIC System?

Under these conditions, the RCIC system is INOPERABLE _____(1)_____.

(2) In addition to restoring the RCIC System to OPERABLE status, what are the MINIMUM actions REQUIRED in accordance with Tech Spec 3.5.3.RCIC System?

A.	(1) AND unavailable (2) Perform CPS 9051.01 HPCS Pump & HPCS Water Leg Pump Operability.

B.	(1) BUT available (2) Perform CPS 9051.01 HPCS Pump & HPCS Water Leg Pump Operability.

C.	(1) AND unavailable (2) Examine logs or other information to determine HPCS system operability.

D.	(1) BUT available (2) Examine logs or other information to determine HPCS system operability.

Answer:	C

Answer Justification / Plausibility Statements

C is correct:

Per ER-AA-310 Implementation of the Maintenance Rule, the terms AVAILABLE and UNAVAILABLE are defined as follows:

- Available - An SSC that is capable of performing its intended function(s). An SSC that can be restored to functional status within the timeframe associated with realistic conditions using proceduralized operator actions.
- Unavailability - The condition where equipment is out-of-service (e.g. tagged out) for corrective or preventive maintenance, or can not perform its Maintenance Rule function.

Per CPS 4201.01C001 Loss of 125VDC MCC1A (1DC13E) Load Impact List, a loss of DC MCC 1A renders the RCIC System not available for normal shutdown without condenser.

Per ITS 3.5.3 RCIC System Condition A, If RCIC is inoperable, A.1 requires HPCS to be verified OPERABLE by administrative means.

Per ITS B3.5.3, OPERABILITY of HPCS is therefore verified within 1 hour when the RCIC System is inoperable. This may be performed as an administrative check, by examining logs or other information, to determine if the HPCS is out of service for maintenance or other reasons. Verification does not require performing the Surveillances needed to demonstrate the OPERABILITY of the HPCS System.

Incorrect Responses:

A is incorrect but plausible. The operability of ITS required systems is normally verified by performing the associated ITS surveillance. Incorrect in this scenario because the ITS bases lists the actions required to comply with ITS 3.5.3 Required Action A.1.

B is incorrect but plausible:

- Part 1 is incorrect because RCIC cannot perform its intended function with a loss of DC MCC 1A. Part 1 is plausible because a loss of AC to RCIC system components will not impact RCIC availability. AC power is supplied to the RCIC primary containment isolation valves. RCIC is fully capable of performing its intended function (injection) without AC power.

- Part 2 is incorrect but plausible because the operability of ITS required systems is normally verified by performing the associated ITS surveillance. Incorrect in this scenario because the ITS bases lists the actions required to comply with ITS 3.5.3 Required Action A.1 and specifically states that the verification does not require performing the surveillances needed to demonstrate the OPERABILITY of the HPCS System.

D is incorrect but plausible because RCIC cannot perform its intended function with a loss of DC MCC 1A. Part 1 is plausible because a loss of AC to RCIC system components will not impact RCIC availability. AC power is supplied to the RCIC primary containment isolation valves. RCIC is fully capable of performing its intended function (injection) without AC power.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.217000	B2.2.40	3.4	4.7	2	1	N/A

System Name
Reactor Core Isolation Cooling System (RCIC)

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):

	Q15/90

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to determine required Technical Specifications actions to answer the question.		
SRO Only Justification Statement:	The determination of SSC availability/unavailability is an SRO function at CPS and is supported by SRO only task 115112.01 Apply the administrative requirements of ON-LINE RISK ASSESSMENT. In addition the administrative actions required to verify HPCS system operability is contained in the Tech Spec Bases, which is SRO only knowledge. 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 must recall facts and apply them to a specific situation.		
NRC Exams Only (as applicable)			
Question Type:	Bank (Duplicate of CL-ILT-N15090) (previous)	**Difficulty:**	N/A
Technical Reference and Revision #:	• ITS 3.5.3 Amend. 163 • ITS B3.5.3 Rev 10-3 • CPS 4201.01C001 Rev. 1 • ER-AA-310 Rev. 11		
Training Objective:	217000.14 Given REACTOR CORE ISOLATION COOLING (RI) System operability status and a copy of Tech Specs, DISCUSS the bases for the REACTOR CORE ISOLATION COOLING (RI) System Tech Spec LCO, related safety limits and Limiting Safety System Settings.		
Previous NRC Exam Use:	ILT 15-1 NRC Exam		

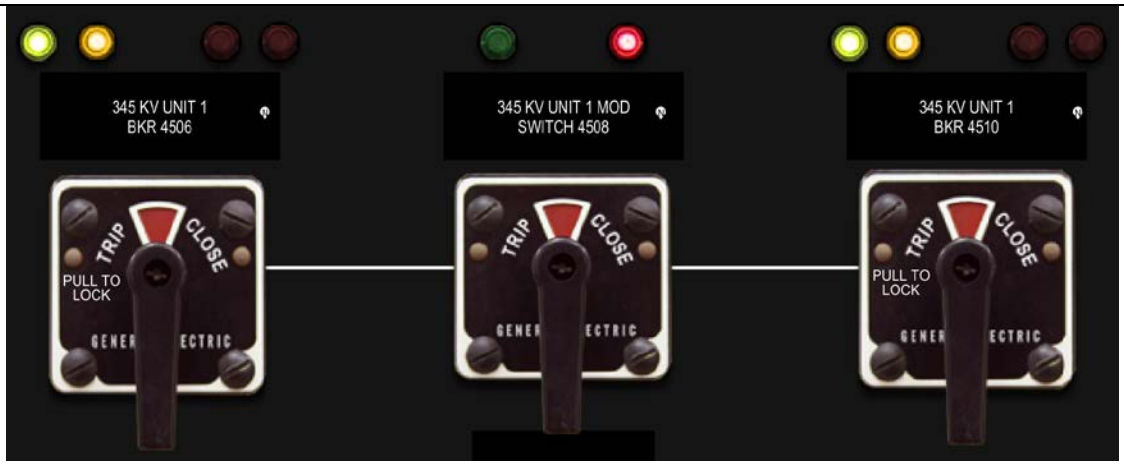
16	ID: 2033329	Points: 1.00

The plant was operating with normal system alignments at 25% power when annunciator 5007-1B TURB TRIP EHC SYS was received.

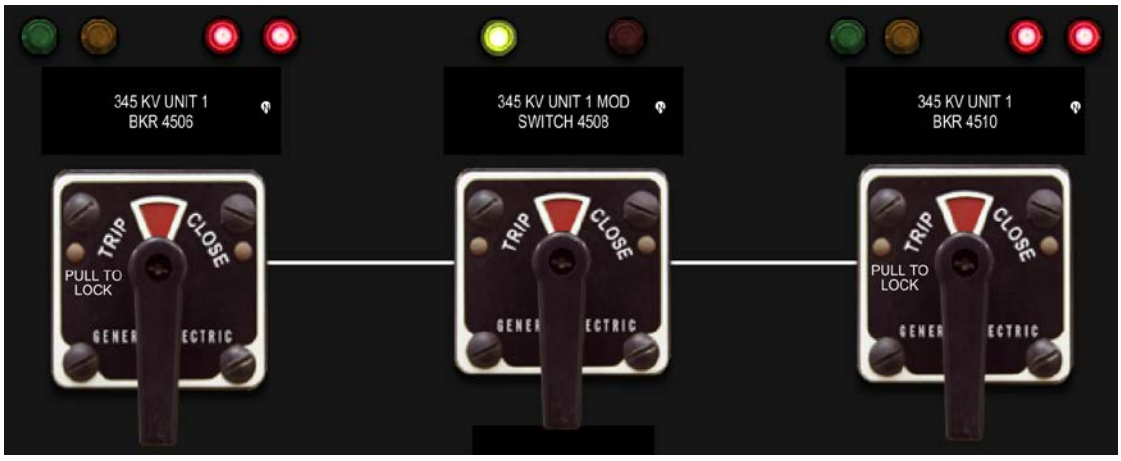
(1) Which of the following graphics (graphic 1 or graphic 2) represents the expected status of GCB 4506, GCB 4510, and MOD 4508 five minutes after receipt of annunciator 5007-1B?

Required actions are to _____(2)_____.

(1)



(2)



A.	(1) Graphic 1 (2) insert control rods until power is < 21.6%
B.	(1) Graphic 2 (2) insert control rods until power is < 21.6%
C.	(1) Graphic 1 (2) secure one of the three operating Circulating Water Pumps
D.	(1) Graphic 2 (2) secure one of the three operating Circulating Water Pumps

EXAMINATION ANSWER KEY

ILT 17-1 NRC SRO Written Exam

Answer:	A

Answer Justification / Plausibility Statements

A is correct:

Per CPS 3004.01 Turbine Startup and Generator Synchronization, the Main Generator is synchronized to the grid between 15% and 18% reactor power, closing GCBs 4510 and 4506.

Per CPS 5007-1B Turb Trip EHC Sys, a turbine trip will cause the generator to trip on reverse power, causing GCBs 4506 and 4510 to open (Graphic 1).

Per CPS 5004.03 (3D), the RPS trip on TSV/TCV closure is bypassed when < 33.3% RTP. At 25% power, the turbine will trip, but the reactor will NOT scram.

Per CPS 4005.01 Loss of Feedwater Heating step 4.5, if the main turbine trips without a SCRAM, then insert control rods to reduce power to < 21.6%.

Incorrect Responses:

B is incorrect but plausible. Opening 4508 would disconnect the Main Generator from the grid, similar to how the RAT transformers are disconnected on a RAT trip, where the RAT Circuit Switcher 4538 opens.

C is incorrect but plausible. CPS 4100.01 Scram, section 4.1 Electric Concerns Post Scram, step 4.1.2 directs securing 1 of the 3 CW pumps if operating with two Reactor Recirc pumps, three CW pumps and the MDRFP on RAT 'A'. Since the stem states that reactor power is at 25%, both RR Pumps are operating in slow speed powered from 4160V Buses 1A and 1B via the Unit Auxiliary Transformers (UATs), so the requirement to secure a Circulating Water Pump does not exist.

D is incorrect but plausible:

- Opening 4508 would disconnect the Main Generator from the grid, similar to how the RAT transformers are disconnected on a RAT trip, where the RAT Circuit Switcher 4538 opens.
- CPS 4100.01 Scram, section 4.1 Electric Concerns Post Scram, step 4.1.2 directs securing 1 of the 3 CW pumps if operating with two Reactor Recirc pumps, three CW pumps and the MDRFP on RAT 'A'. Since the stem states that reactor power is at 25%, both RR Pumps are operating in slow speed powered from 4160V Buses 1A and 1B via the Unit Auxiliary Transformers (UATs), so the requirement to secure a Circulating Water Pump does not exist.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
245000.A2.01	A2.01	3.7	3.9	2		4

System Name
Main Turbine Generator and Auxiliary Systems

Category Statement
Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS ; 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
Turbine trip

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

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee must demonstrate the ability to predict the impact of a Turbine trip on the Main Turbine Generator and Auxiliary System controls by choosing a graphic that shows the predicted alignment following a turbine trip to answer the question.		
SRO Only Justification Statement:	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 written at the comprehension level (high cog). The examinee must recognize the interaction between systems and determine consequences/implications. (2-RI)		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 3004.01 Rev. 33g • CPS 5007.01 (1B) Rev. 30c • CPS 5004.03 (3D) Rev. 28b • CPS 4005.01 Rev. 18e • CPS 4100.01 Rev. 23c		
Training Objective:	(410001.01) From the MCR, complete Control Room Actions to respond to a Reactor Scram, IAW CPS 4100.01 Reactor Scram.		
Previous NRC Exam Use:	None		

17	ID: 2048307	Points: 1.00

The plant is in Mode 4.

4160V Bus 1B1 is energized from the Reserve Auxiliary Transformer (RAT).

CPS 9080.22 DIESEL GENERATOR 1B - ECCS INTEGRATED surveillance is in progress

The breaker for 1FC02PB Fuel Pool Clg Pump 1B failed to trip when the Loss of Power and ECCS initiation logic circuits were actuated.

Which of the following systems must be declared INOPERABLE?

A.	Div 2 DC Sources
B.	Div 2 Diesel Generator
C.	Div 2 AC Power Distribution System
D.	Reserve Auxiliary Transformer (RAT)

Answer:	B

Answer Justification / Plausibility Statements
B is correct: Per ITS B3.8.1 AC Sources - Operating, proper sequencing of loads, **including tripping of nonessential loads**, is a required function of DG 1A and DG 1B OPERABILITY. Per CPS 9080.22 Diesel Generator 1B - ECCS Integrated, step 8.1.36.1, 1FC02PB breaker is verified tripped when the Div 2 ECCS logic is initiated. This step partially satisfies ITS SR 3.8.1.19 (ref. 9080.22 step 9.1.1.12.2). This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ECCS initiation signal. If the trip is inoperable, the associated Diesel Generator must be declared INOPERABLE. Incorrect Responses: A is incorrect but plausible. Control power for the 1FC02PB breaker is supplied from DC MCC 1B. Incorrect because a failure of the breaker to shunt trip does not impact Div 2 DC Sources, which is considered operable when energized from an operable battery and battery charger. C is incorrect but plausible. 1FC02PB is powered from the Div 2 AC Power Distribution System. Incorrect because operability of the Div 2 AC Power Distribution System is established when the correct breaker alignments and voltage to required AC bus electrical power distribution subsystems exists. D is incorrect but plausible since the stem states that 4160V Bus 1B1 is energized from the RAT and would be correct if the operability of the associated off-site source was impacted by operability of shunt tripped loads. Since the off-site sources are disconnected from the bus during LOP/LOCA testing, shunt tripping of selected loads does not impact the respective off-site source.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
GS.233000	B2.1.07	4.4	4.7	2	2	N/A

System Name
Fuel Pool Cooling and Clean-up

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

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):

	Q17/92

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to demonstrate the ability to evaluate Fuel Pool Cooling and Cleanup system performance during a surveillance test to make operational judgments concerning implementation of Technical Specification requirements.		
SRO Only Justification Statement:	This question is linked to SRO only task 140109.23 Apply The Administrative Requirements For Execution Of Technical Specifications Or Off-Site Dose Calculation Manual Requirements and 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases. This question also requires knowledge of TS bases for Diesel Generator operability		
Additional Information:	This is a low cog question written at the memory level. The examinee has to recall DG operability requirements per ITS 3.8.1 AC Sources - Operating (bases) to answer the question.		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• ITS B3.8.1 (B 3.8-30) Revision No. 14-4 • CPS 9080.22 Rev. 34d		
Training Objective:	262001.12 Given Auxiliary Power 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: 2033275	Points: 1.00

The plant was operating at rated thermal power (RTP).

At 1500 a transient occurred that resulted in a SCRAM and loss of the Reserve Auxiliary Transformer (RAT).

At 1502 the immediate actions for the scram were completed and CPS 4200.01 Loss of AC Power was entered.

Which of the following subsequent actions will be given the highest priority?

A.	Bypass the Condensate Polishers.
B.	Restore RAT supply to the 4.16 KV Busses 1A1, 1B1, and 1C1.
C.	Initiate Reactor Recirc Pump Auxiliary Seal Injection Pump operation.
D.	Verify all rods inserted at the Rod Action Control System (RACs) panels.

Answer:	C

Answer Justification / Plausibility Statements

C is correct:

Per CPS 4200.01 Loss of AC Power, SBO: Technical Bases Summary, states that system leakage for a SBO, including RR pump seals (~ 38 gpm) and inventory loss from SRVs, it is assumed to be ~ 100 gpm.

The commonality between a SBO and loss of the non-ECCS buses is that both events will result in undervoltage trips of the CRD Pumps, a loss of the RR Pumps, and a loss of the CCW pumps.

Per EP-AA-1003 Addendum 3 Emergency Action Levels For Clinton Station basis for MU6 (CL 2-92), a stuck open SRV or SRV leakage is not considered either identified or unidentified leakage by Technical Specification and, therefore, is not applicable to the MU6 EAL.

Therefore, leakage past the RR Pump seals will result in unidentified leakage in the drywell of > 10 gpm for ≥ 15 minutes.

Incorrect Responses:

A is incorrect but plausible. The conditions in the stem (scram and loss of non-vital power) will result in a loss of vacuum with no ability for mitigation with no power available to the Condenser Vacuum Pumps. CPS 4004.02 Loss of Vacuum directs bypassing the Condensate Polishers to prevent overheating the resin during degrading vacuum conditions. This action would not be prioritized however, because of the loss of non-vital power to 1CD016 Condensate Polisher Bypass Valve and to the Condensate/Condensate Booster System Pumps.

B is incorrect but plausible. Emergency Action Level (EAL) MA1 is exceeded if AC power capability to the emergency busses is reduced to a single power source for 15 minutes or longer such that an additional single failure would result in a station blackout. Under the conditions provided in the stem, however, the ERAT and DGs are available to the safety busses, so MA1 is not imminently threatened from a loss of the RAT transformers and restoring the RAT supply to the safety busses would be prioritized lower than initiating RR Aux Seal Injection pump operation.

D is incorrect but plausible. Failure of RPS to shutdown the reactor can result in exceeding the threshold for EAL MS3, MA3 or MU3 depending on the severity of the ATWS. The threshold values for these EALs, however, are based on reactor power and not on the ability to determine if shutdown criteria is met. Furthermore, since the immediate actions for the scram are complete (rods have been verified inserted) and the loss of non-vital AC will not preclude the operator from determining rod positions at P680 - verifying rod positions at the RACS panels is not required.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
201001.A2.04	A2.04	3.8	3.9*	2		1

System Name
Control Rod Drive Hydraulic System

Category Statement
Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC 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
†Scram conditions

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

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to determine the impact of the loss of BOP power and resultant loss of CRD cooling water flow to the RR Pump seals and then determine that the MU6 threshold values will be exceeded requiring actions to be taken per the Emergency Plan.		
SRO Only Justification Statement:	This question is linked to SRO only task 997777.03 Emergency Plan Activities performed by an SRO 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 at the analysis and comprehension level because the examinee has to analyze the conditions in the stem and then determine required actions based on that analysis (3-SPK/SPR).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• EP-AA-1003 Addendum 3 (CL 2-4, 2-5, 2-6, 2-91,and 2-92) Rev. 2 • CPS 4200.01 Rev. 25		
Training Objective:	201001.09 DISCUSS the effect: a. A total loss or malfunction of the Control Rod Drive Hydraulic System has on the plant. b. A total loss or malfunction of various plant systems has on the Control Rod Drive Hydraulic System.		
Previous NRC Exam Use:	None		

19	ID: 1799174	Points: 1.00

In order to maintain an ACTIVE status, a Senior Reactor Operator (NOT qualified as a Shift Manager) must actively stand watch as the _____(1)_____ for a MINIMUM of _____(2)_____ hours per calendar quarter.

A.	(1) Control Room Supervisor (CRS) ONLY (2) 56

B.	(1) Control Room Supervisor (CRS) ONLY (2) 40

C.	(1) Control Room Supervisor (CRS) OR Work Execution Coordinator (WEC) (2) 56

D.	(1) Control Room Supervisor (CRS) OR Work Execution Coordinator (WEC) (2) 40

Answer:	A

Answer Justification / Plausibility Statements
A is correct: Per OP-AA-105-102 NRC Active License Maintenance step 4.1.1 - MAINTAIN an active license by actively performing the functions of RO, or SRO. Per OP-AA-105-102 NRC Active License Maintenance step 4.1.1.2 - an active SRO license is maintained by performing the duties of **Shift Manager or Unit Supervisor** for a minimum of **seven 8-hour** or five 12-hour **shifts per calendar quarter**, including turnover to the next shift. Incorrect Responses: B is incorrect but plausible. The **reactivation** requirement for an SRO license to return to an "active status" is 40 hours of shift functions in the presence and under the sole direct supervision of an active SRO. C is incorrect but plausible. The WEC position is filled by an SRO, but WEC watches do not count for active license maintenance purposes. D is incorrect but plausible: • The WEC position is filled by an SRO, but the WEC watches do not count for active license maintenance purposes. • The **reactivation** requirement for an SRO license to return to an "active status" is 40 hours of shift functions in the presence and under the sole direct supervision of an active SRO.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.1.04	B2.1.04	3.3	3.8	3	N/A	N/A

System Name
Conduct of Operations

Category Statement
Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, □no-solo□ operation, maintenance of active license status, 10CFR55, etc. (CFR: 41.10 / 43.2)

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):

	Q19/94

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee is required to be knowledgeable of the requirements for maintaining an active license in order to answer the question.		
SRO Only Justification Statement:	This question requires knowledge of the requirements of maintaining an active SRO license.		
Additional Information:	This question is a low cog question written at the memory level. The examinee must recall facts from a procedure to answer the question (1-F/1-P).		
NRC Exams Only (as applicable)			
Question Type:	Bank (duplicate of CL-ILT-N12098)	**Difficulty:**	N/A
Technical Reference and Revision #:	OP-AA-105-102 Rev. 13		
Training Objective:	LP85801.2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no solo” operation, maintenance of active license status, 10CFR55, etc.		
Previous NRC Exam Use:	ILT 12-1 NRC Exam		

20	ID: 2035843	Points: 1.00

A refueling outage is in progress.

A hydrostatic test of the RPV is scheduled to be performed IAW CPS 9059.01 Reactor Coolant System Leakage Test.

To conduct this evolution, what type of pre-job briefing is required?

A.	tailored pre-job brief
B.	standard pre-job brief
C.	Infrequent Plant Activity (IPA) brief
D.	Heightened Level of Awareness (HLA) brief

Answer:	C

Answer Justification / Plausibility Statements
C is correct: Per OP-AA-108-110 Evaluation of Special Tests or Evolutions, a Reactor Pressure Vessel Hydro is an example of a special, infrequently performed surveillance test involving complicated sequencing. Per HU-AA-1211 Pre-Job Briefings activities with the potential to degrade the plant level of nuclear safety such as, Special Tests (determined using OP-AA-108-110) require an Infrequent Plant Activity (IPA) briefing. Incorrect Responses: A is incorrect but plausible. A tailored pre-job brief used for **high-risk** and simple or frequent work activities. Performance of CPS 9059.01 is neither simple or frequently performed. B is incorrect but plausible. A standard pre-job briefing is used for low risk jobs requiring coordination of two or more people or departments. CPS 9059.01 does require coordination of two or more people or departments, but is not a low risk activity. D is incorrect but plausible. An HLA briefing is used for activities that pose additional risk on the plant, but are not appropriate for evolutions that are seldom performed that have the potential to degrade the plant level of nuclear safety (such as performing a vessel hydro). The IPA brief requires Senior Line Management involvement to provide emphasis on the important nature of the evolution to be performed.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.2.07	B2.2.07	2.9	3.6	3	N/A	N/A

System Name
Equipment Control

Category Statement
Knowledge of the process for conducting special or infrequent tests. (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

EXAMINATION ANSWER KEY

ILT 17-1 NRC SRO 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):

	Q20/95

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to have knowledge of the process for conducting special tests by determining the type of briefing required to answer the question.		
SRO Only Justification Statement:	This question is tied to SRO only task 101102.01 Apply the administrative requirements of IMPLEMENTATION AND CONTROL OF SURVEILLANCE TESTING. The Shift Manager is also listed as having responsibility for the IPA briefing IAW HU-AA-1211-F-03.		
Additional Information:	This question is a low cog question written at the memory level. The examinee must recall facts from a procedure to answer the question (1-F/1-P).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• HU-AA-1211 Rev. 11 • OP-AA-108-110 Rev. 3		
Training Objective:	LP85802.2.2.7 Knowledge of the process for conducting special or infrequent tests.		
Previous NRC Exam Use:	None		

21	ID: 1799173	Points: 1.00

Due to a station emergency, the Shift Manager has assumed command and control as the Shift Emergency Director.

The Shift Emergency Director must authorize emergency radiation exposures in excess of _____(1)_____, and _____(2)_____ authorize the use of Potassium Iodide (KI) if a potential of high thyroid exposure to radioactive iodine exists.

A.	(1) 2 Rem (2) may

B.	(1) 5 Rem (2) may

C.	(1) 2 Rem (2) may NOT

D.	(1) 5 Rem (2) may NOT

Answer:	B

Answer Justification / Plausibility Statements

B 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.

Incorrect Responses:

A is incorrect but plausible. Per RP-AA-203 Exposure Control and Authorization, step 4.1.2, 2000 mrem TEDE/yr is the administrative dose control limit. RP-AA-203 provides the normal process for raising the admin dose control level, but these directions do not apply during E-Plan implementation.

C is incorrect but plausible. RP-AA-203 Exposure Control and Authorization does provides the normal process for raising the admin dose control level above 2 Rem, but these directions do not apply during E-Plan implementation. EP-AA-112-200-F-01 lists the non-delegable responsibilities of the Station Emergency Director, which includes KI issuance, which means that the SED retains responsibility for KI issuance after command and control is transferred to the EOF.

D is incorrect but plausible. EP-AA-112-200-F-01 lists the non-delegable responsibilities of the Station Emergency Director, which includes KI issuance, which means that the SED retains responsibility for KI issuance after command and control is transferred to the EOF.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.3.04	B2.3.04	3.2	3.7	3	N/A	N/A

System Name
Radiation Control

Category Statement
Knowledge of radiation exposure limits under normal or emergency conditions. (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):

	Q21/96

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to have knowledge of the emergency exposure limits to answer the question.		
SRO Only Justification Statement:	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:	This question is a low cog question written at the memory level. The examinee must recall procedure steps to answer the question (1-F/1-P).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• EP-AA-113 Rev. 13 • RP-AA-203 Rev. 5		
Training Objective:	LP85803.2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.		
Previous NRC Exam Use:	None		

22	ID: 1799529	Points: 1.00

The unit was operating at rated thermal power when an unisolable leak occurred on the RHR Pump 'A' suction line resulting in annunciator 5004-3F SPDS CSF alarming.

The Shift Technical Advisor (STA) recommends entering CPS 4304.01 FLOODING.

The Control Room Supervisor (CRS) _____(1)_____ implement CPS 4304.01 FLOODING because _____(2)_____.

A.	(1) will (2) under current conditions, CPS 4304.01 FLOODING is the highest tier procedure

B.	(1) will (2) off-normal procedures may be implemented concurrently with higher tier procedures

C.	(1) will NOT (2) off-normal procedures can NOT be performed concurrently with higher tier procedures

D.	(1) will NOT (2) under current conditions, CPS 4304.01 FLOODING contains steps that conflict with higher tier procedures

Answer:	B

Answer Justification / Plausibility Statements

B is correct.

Per CPS 1005.09 EMERGENCY OPERATING PROCEDURE (EOP) AND SEVERE ACCIDENT GUIDELINE (SAG) PROGRAM, step 8.12.1 Hierarchy of Emergency Response Procedures list the hierarchy relationship of CPS procedures applicable to performance of emergency response activities as follows:

1. SAG Flowcharts/SAG Support Procedures
2. EOP Flowcharts/EOP Support Procedures
3. Off-Normal Response Procedures
4. Annunciator Response Procedure actions
5. ABNORMAL OPERATING section actions located in the System Operating Procedures
6. Integrated Plant Operating Procedures
7. NORMAL/INFREQUENT OPERATING section actions located in the System Operating Procedures
8. All other CPS procedures.

An unisolable leak on the RHR 'A' Pump suction line will result in Suppression Pool leakage into the Secondary Containment requiring entry into EOP-8 Secondary Containment Control and EOP-6 Primary Containment Control (#2 on the list of procedures above). CPS 4304.01 FLOODING is an off-normal response procedure, and is ranked #3 on the list of procedures above.

Per 1005.09 step 8.12.2.3, as time/resources permit, a review of the lower tier procedure(s) should be performed to verify all applicable actions have been or will be performed.

Incorrect Responses:

A is incorrect but plausible. This answer would be correct if CPS 4304.01 was the only procedure requiring entry, and is therefore the highest tier procedure with actions that are required to be performed.

C is incorrect but plausible. This answer would be correct if Off-Normal and Emergency Operating Procedures (EOPs) were treated the same as EOPs and Severe Accident Guidelines (SAGs). The EOPs direct exiting <u>all</u> EOPs when entering the SAGs.

D is incorrect but plausible. This answer would be correct if CPS 4304.01 contained steps that conflicted with actions in EOP-6 and/or EOP-8. Per CPS 1005.09 step 8.12.1.3, should a conflict in operating direction exist between the hierarchy of emergency response procedures, then the upper tier procedure in use shall take precedence.

<u>K/A Data</u>

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.4.05	B2.4.05	3.7	4.3	3	N/A	N/A

System Name
Emergency Procedures /Plan

Category Statement
Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions. (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):

	Q22/97

Other NRC Data

References Provided:	None		
K/A Justification Statement:	Question meets the KA because the examinee is required to have knowledge of the organization of the operating procedure network (hierarchy/usage/compliance) for normal, abnormal and emergency evolutions to answer this question.		
SRO Only Justification Statement:	This question requires the examinee to have detailed knowledge of procedure content and 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 written at the analysis/comprehension level (high cog). The examinee has to analyze conditions in the stem and solve a problem using knowledge to answer the question (3-SPK).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• CPS 1005.09 Rev. 10b • CPS 4304.01 Rev. 6c • CPS 4402.01 Rev. 30 • CPS 4406.01 Rev. 30		
Training Objective:	LP85804.2.4.5 Knowledge of the organization of the operating procedures network for normal / abnormal / and emergency evolutions.		
Previous NRC Exam Use:	None		

23	ID: 1799172	Points: 1.00

Core Alterations were stopped by the Refuel SRO due to plant conditions.

Who has the authority to grant permission to resume fuel movement?

A.	Refuel SRO
B.	Shift Manager
C.	Lead Refuel SRO
D.	Control Room Supervisor

Answer:	B

Answer Justification / Plausibility Statements
B is correct - Per CPS 3703.01 Core Alterations Section 4.29, the Shift Manager's permission is required to resume fuel movement. Incorrect Responses: A is incorrect but plausible. The Refuel SRO has a number of responsibilities during core alterations, but does not possess the authority to grant permission to resume fuel movement. C is incorrect but plausible. The Lead Refuel SRO has responsibilities during core alterations but does not possess the authority to grant permission to resume fuel movement. D is incorrect but plausible. The Control Room Supervisor has responsibilities during core alterations but does not possess the authority to grant permission to resume fuel movement.

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.1.35	B2.1.35	2.2	3.9	3	N/A	N/A

System Name

Category Statement
Knowledge of the fuel-handling responsibilities of SROs. (CFR: 41.10 / 43.7)

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
43.7	43.7

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

Other NRC Data

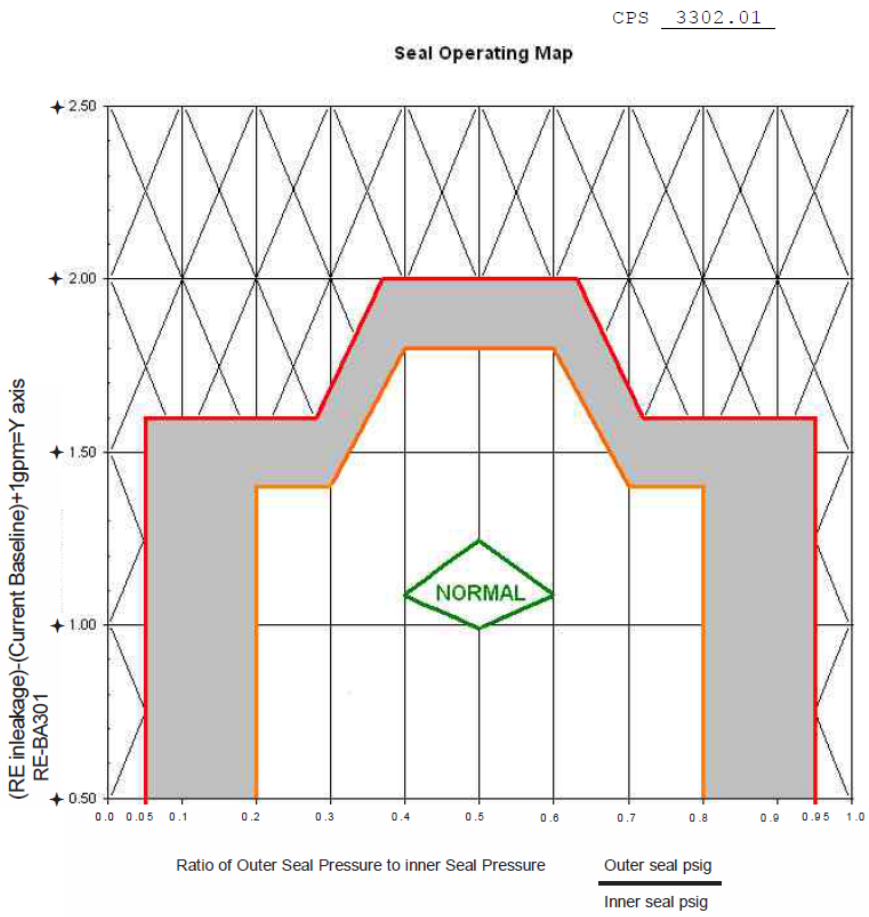
References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to identify the actions required of the Refueling SRO in the event that core alts are impaired due to plant conditions.		
SRO Only Justification Statement:	This question is linked to 10CFR55.43(b)(6) - Procedures and limitations involved in core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity. This question also tests knowledge of SRO-only responsibilities during refueling operations.		
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 (as applicable)			
Question Type:	Bank (duplicate of CL-ILT-N11093)	**Difficulty:**	N/A
Technical Reference and Revision #:	CPS 3703.01 Rev. 28		
Training Objective:	LP85802.2.2.26 Knowledge of refueling administrative requirements.		
Previous NRC Exam Use:	ILT 10-1 NRC Exam		

24	ID: 1799171	Points: 1.00

The plant is operating at rated thermal power.

RR Pump 'A' seal parameters indicate a slowly degrading upper pump seal.

Given the following and assuming RR Pump 'A' lower seal indicated 1054 psig when the degrading trend was identified, an orderly plant shutdown due to a FAILED upper RR Pump 'A' seal will be required when RR Pump 'A' upper seal pressure reaches _____ psig (assume a Y-axis value of 1.00 gpm).



Operation in Shaded Area = significantly degraded seal, initiate an Issue Report and schedule shutdown for repair.

Operation Outside Shaded Area = failed seal, initiate an Issue Report and notify station management. Prepare to commence normal plant shutdown.

A.	1012

B.	854

C.	221

D.	42

Answer:	D

Answer Justification / Plausibility Statements
D is correct: • CPS 3302.01 step 8.3.1.3.2 provides directions for determining RR Pump Seal conditions. A failure of the upper seal will result in upper seal pressure lowering. • CPS 3302.01 step 8.3.1.5 and the seal operating map states that if the outer to inner seal pressure ratio is outside the shaded area of the seal operating map, then commence an orderly plant shutdown per CPS 3006.01 Unit Shutdown. For an upper seal failure (lowering outer seal pressure), a ratio of outer seal psig to inner seal psig = 0.05 indicates an upper seal failure. Since lower seal pressure in the stem is 1054 psig, upper seal failure is indicated when upper seal pressure = 42 psig (1054 * .05 = 52.7 psig) Incorrect Responses: A is incorrect but plausible. A lower seal failure is indicated by the rising value for the ratio of outer to inner seal pressure. A ratio of 0.95 on the Seal Operating Map indicates a failed lower seal, requiring an orderly plant shutdown to be commenced per CPS 3006.01 Unit Shutdown. 1054 * 0.95 = 1001.3 psig B is incorrect but plausible. A lower seal failure is indicated by the rising value for the ratio of outer to inner seal pressure. A ratio of 0.8 on the Seal Operating Map indicates a significantly degraded lower seal, requiring a shutdown to be scheduled but not commenced. 1054 * 0.8 = 843.2 psig C is incorrect but plausible. Operation in the shaded area indicates a significantly degraded seal (not a failed seal) and requires a shutdown to be scheduled but not commenced. 1054 * 0.2 = 210.8 psig

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.4.47	B2.4.47	4.2	4.2	3	N/A	N/A

System Name
Emergency Procedures /Plan

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):

	Q24/99

Other NRC Data

References Provided:	None		
K/A Justification Statement:	This question meets the KA because the examinee has to diagnose the trend in seal parameters provided in the stem and then determine when a plant shutdown is required based on that analysis to answer the question.		
SRO Only Justification Statement:	This is an SRO only question because the SRO position determines when a plant shutdown is required when RR Pump seal parameters reach their failure point. 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:	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 (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	CPS 3302.01 Rev. 36c		
Training Objective:	202001.11 EVALUATE given key Reactor Recirculation System parameters, if needed DETERMINE a course of action to correct or mitigate the following abnormal condition(s):		
Previous NRC Exam Use:	None		

25	ID: 2035852	Points: 1.00

At 0000 the plant was operating at rated thermal power with the following conditions:

- Div 1 DG INOPERABLE
- appropriate LCOs entered

THEN, at 0845, annunciator 5065-8B RHR B Out Of Service was received due to a motor breaker control power failure.

Which of the following cases describe the LCO Condition(s) that is (are) required to be entered?

Case	Time	Condition
1	0000	Enter ITS 3.6.1.7 Condition A - One RHR containment spray subsystem inoperable.
2	0400	Enter ITS 3.6.1.7 Condition A - One RHR containment spray subsystem inoperable.
3	0845 0845	Enter ITS 3.6.1.7 Condition A - One RHR containment spray subsystem inoperable, AND Enter ITS 3.6.1.7 Condition B - Two RHR containment spray subsystems inoperable.
4	0845 1245	Enter ITS 3.6.1.7 Condition A - One RHR containment spray subsystem inoperable, AND Enter ITS 3.6.1.7 Condition B - Two RHR containment spray subsystems inoperable.

Case...

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

Answer:	D

Answer Justification / Plausibility Statements

D is correct:

Per ITS 3.8.1 AC Sources - Operating, with the Div 1 DG inop and RHR 'B' inop, LCO 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, with a completion time of 4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s).

Since the RHR B motor breaker control power failure was identified at 0845, ITS LCO 3.6.1.7 Condition A would be entered due to inoperability of the B loop of Containment Spray at 0845.

Additionally, since the concurrent condition occurred at 0845, the completion time for 3.8.1 B.2 is 1245 for the following LCOs:

- 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 based on the definition of OPERABILITY. RHR Containment Spray (the supported system) requires the associated DG (the support system) for OPERABILITY. ITS 3.0.6 however, does not require entry into the Conditions and Required Actions for a supported system due solely to inoperability of the support system.

B is incorrect but plausible. ITS 3.8.1.B.2 requires declaring required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable 4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s). 0000 + 4 hours = 0400. Incorrect because the redundant required feature(s) (RHR 'B') is operable at 0000 and ITS 3.8.1.B.2 is not required.

C is incorrect but plausible. ITS 3.8.1.B.2 requires declaring required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable 4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s). The inoperability of RHR 'B' occurs at 0845. Incorrect because 3.8.1.B.2 provides for a 4 hour completion time from the time of inoperability (0845 + 4 hours = 1245).

K/A Data

K/A Number	Viewed K/A	RO Value	SRO Value	Tier	RO/SRO Group	Safety Function
B2.2.23	B2.2.23	3.1	4.6	3	N/A	N/A

System Name
Equipment Control

Category Statement
Ability to track Technical Specification limiting conditions for operations. (CFR: 41.10 / 43.2 / 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
43.2	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):

	Q25/100

Other NRC Data

References Provided:	ITS 3.8.1 page 3.8-1 and 3.8-2 with the following redactions: • everything above the double lines on 3.8-1 except the LCO title • Condition A completion time • Condition B.1 completion time		
K/A Justification Statement:	Question meets the KA because the examinee has to analyze the conditions in the stem and then track the entry time for impacted LCOs to answer the question.		
SRO Only Justification Statement:	This question is linked to 10CFR55.43(b)(2) - Facility operating limitations in the technical specifications and their bases.		
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 determine actions based on that analysis (3-SPK).		
NRC Exams Only (as applicable)			
Question Type:	New	**Difficulty:**	N/A
Technical Reference and Revision #:	• ITS 1.1 (1.0-5) Amendment No. 149 • ITS 3.0.6 (3.0-2) Amendment No 213 • ITS 3.8.1 (3.8-2) Amendment No. 141		
Training Objective:	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.		
Previous NRC Exam Use:	None		