U.S. Nuclear Regulatory Commission

Site-Specific RO Written Examination

Applicant Information

Name:	
Date: July 22, 2010	Facility/Unit: IPEC Unit 2
Region: I ☒ II ☐ III ☐ IV ☐	Reactor Type: W ☒ CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	________ Points
Applicant's Score	________ Points
Applicant's Grade	________ Percent

SUBJECTIVE SCORE INSTRUCTOR USE ONLY

100 90 80 70 60
50 40 30 20 10
9 8 7 6 5
4 3 2 1 0

FORM NO. 888-E

IMPORTANT

USE NO. 2 PENCIL ONLY
• MAKE DARK MARKS
• ERASE COMPLETELY TO CHANGE
• EXAMPLE: A B ■ D E

TO USE SUBJECTIVE SCORE FEATURE:
• Mark total possible subjective points
• Only one mark per line on key
• 163 points maximum

EXAMPLE OF STUDENT SCORE:

PART 1

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NAME: RO ANSWER KEY
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PERIOD

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TEST RECORD	
PART 1	
PART 2	
TOTAL	

	(T)	(F)			KEY
	%	■	3		■
1	A	B	■	D	E
2	■	B	C	D	E
3	A	■	C	D	E
4	■	B	C	D	E
5	A	■	C	D	E
6	A	B	■	D	E
7	A	B	■	D	E
8	■	B	C	D	E
9	■	B	C	D	E
10	A	B	C	■	E
11	A	B	C	■	E
12	A	■	C	D	E
13	■	B	C	D	E
14	■	B	C	D	E
15	A	B	C	■	E
16	A	B	■	D	E
17	■	B	C	D	E
18	A	■	C	D	E
19	A	B	C	■	E
20	A	■	C	D	E
21	A	B	C	■	E
22	■	B	C	D	E
23	A	B	C	■	E
24	A	B	C	■	E
25	A	■	C	D	E
26	A	B	C	■	E
27	■	B	C	D	E
28	A	■	■	D	E
29	A	B	■	D	E
30	A	B	■	D	E
31	A	■	C	D	E
32	A	B	■	D	E
33	A	■	C	D	E
34	A	B	■	D	E
35	A	■	C	D	E
36	A	B	C	■	E
37	A	B	C	■	E
38	A	B	■	D	E
39	A	B	C	■	E
40	A	■	C	D	E
41	A	B	C	■	E
42	■	B	C	D	E
43	A	B	C	■	E
44	■	B	C	D	E
45	A	B	C	■	E
46	A	B	■	D	E
47	A	■	C	D	E
48	A	■	C	D	E
49	A	B	■	D	E
50	A	B	■	D	E

B+C correct #28
per post EXAM Comment
Restulen JLK 9/1/10

FEED THIS DIRECTION

CUSTOMER SERVICE 1-800-SCANTRON

M12 1008 999 12 11 10 9 8

IMPORTANT	
USE NO. 2 PENCIL ONLY	TO USE SUBJECTIVE SCORE FEATURE:
• MAKE DARK MARKS	• Mark total possible subjective points
• ERASE COMPLETELY TO CHANGE	• Only one mark per line on key
• EXAMPLE: A B C D E	• 163 points maximum
	EXAMPLE OF STUDENT SCORE:

PART 2

NAME	RO ANSWER KEY		
SUBJECT		TEST NO.	
DATE		PERIOD	

TEST RECORD	
PART 1	
PART 2	
TOTAL	

	(T)	(F)			KEY
	%	2	3		5
51	A	B	C	D	E
52	A	B	C	D	E
53	A	B	C	D	E
54	A	B	C	D	E
55	A	B	C	D	E
56	A	B	C	D	E
57	A	B	C	D	E
58	A	B	C	D	E
59	A	B	C	D	E
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63	A	B	C	D	E
64	A	B	C	D	E
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66	A	B	C	D	E
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92	A	B	C	D	E
93	A	B	C	D	E
94	A	B	C	D	E
95	A	B	C	D	E
96	A	B	C	D	E
97	A	B	C	D	E
98	A	B	C	D	E
99	A	B	C	D	E
100	A	B	C	D	E

FEED THIS DIRECTION

20

25

8 10054 00013 6

IPEC Unit 2 NRC Written Exam Answer Key
July 12, 2010

1	**C**	26	**D**	51	**B**	76	**A**
2	**A**	27	**A**	52	**C**	77	**C**
3	**B**	28	**B** +C	53	**C**	78	**B**
4	**A**	29	**C**	54	**B**	79	**A**
5	**B**	30	**C**	55	**D**	80	**A**
6	**C**	31	**B**	56	**A**	81	**B**
7	**C**	32	**C**	57	**C**	82	**D**
8	**A**	33	**B**	58	**A**	83	**A**
9	**A**	34	**C**	59	**A**	84	**D**
10	**D**	35	**B**	60	**C**	85	**B**
11	**D**	36	**D**	61	**B**	86	**B** +C
12	**B**	37	**D**	62	**D**	87	**B**
13	**A**	38	**C**	63	**D**	88	**A**
14	**A**	39	**D**	64	**D**	89	**A**
15	**D**	40	**B**	65	**B**	90	**C**
16	**C**	41	**D**	66	**B**	91	**B**
17	**A**	42	**A**	67	**A**	92	**B**
18	**B**	43	**D**	68	**B**	93	**B**
19	**D**	44	**A**	69	**C**	94	**C**
20	**B**	45	**D**	70	**A**	95	**D**
21	**D**	46	**C**	71	**B**	96	**B**
22	**A**	47	**B**	72	**D**	97	**B**
23	**D**	48	**B**	73	**C**	98	**C**
24	**D**	49	**C**	74	**A**	99	**B**
25	**B**	50	**C**	75	**C**	100	**A**

NOTE! FOR QUESTIONS 28 & 86
2 correct answers
are BEING ACCEPTED
POST EXAM RESOLUTION
9/1/10

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000007K102 Knowledge of the operational implications of the following concepts as they apply to the reactor trip: - Shutdown margin	
	Importance	3.4	

Question # 1

Which of the following statements is correct regarding boration to required concentration following a reactor trip from Mode 1 conditions? (Assume Tavg is stable at 547^0F)

A. Boration does not have to be considered because the online boron concentration is greater than the required shutdown boron concentration.

B. Boration to the required shutdown concentration may be delayed up to 8 hours if reactor power had been less than 50% for the 48 hours prior to the trip.

C. Boration to the required shutdown concentration may be delayed up to 8 hours if reactor power had been greater than 50% for the 48 hours prior to the trip.

D. Boration to the required shutdown concentration must be commenced without delay regardless or power level prior to the trip.

Answer: C

Explanation/Justification:

A. Incorrect but plausible because the online boron concentration often does exceed the required shutdown concentration.
B. Incorrect but plausible because an operator may be confused as to which power level allows for delay.
C. Correct.
D. Incorrect but plausible because an operator may assume no allowance for Xenon is allow in boration requirements.

Technical References: 2-POP-3.2

Proposed References to be provided: None

Learning Objective:		I2LP-ILO-EOPS01 5	
Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	
Question History:	Last 2 NRC Exam2 at IPEC:		NA
Question Cognitive Level:	Memory or Fundamental Knowledge:		X
	Comprehension or Analysis:		
10 CFR Part 55 Content:	55.41	(b) 5	
	55.43	(b)	
Comments:			

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000025K301 Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal System: - Shift to alternate flowpath	
	Importance	3.1	

Question # 2

Given the following:

- The plant is at reduced inventory in preparation for vacuum fill of the RCS following a mid-cycle RCP seal replacement.
- The RCS is vented with the Pressurizer manway removed.
- An RCS leak occurred
- RCS level is decreasing
- RHR flow is oscillating
- The operating RHR pump was subsequently secured.
- The crew is performing actions per AOP-RHR-1, Loss of RHR.

Which of the following identifies the initial desired position of HCV 638, 21 RHR HX and HCV 640, 22 RHR HX and why?

A. HCV 638 and 640 are left open to provide a gravity feed path from RWST

B. HCV 638 and 640 are left open in preparation for restarting the pump from RWST

C. HCV 638 and 640 are closed in preparation for restarting the pump

D. HCV 638 and 640 are closed in an attempt to isolate the leak

Answer: A

Explanation/Justification:

A. Correct. If charging pumps cannot maintain RCS inventory, or a more rapid increase in level is desired, the RWST is gravity drained to the RCS via RHR. These valves must be left open.

B. Incorrect. Plausible because aligning the RHR to the RWST would provide subcooled water to the suction and the capacity of the RHR pumps would increase RCS inventory more rapidly. The procedure does not direct this action.

C. Incorrect. Plausible because for conditions when the pump trips or is stopped for reasons other than cavitation, HCV 638 and 640 are closed in preparation for starting an RHR pump.

D. Incorrect. Plausible because closing these valves may isolate the leak; however, the procedure leave them open to provide a gravity flow path from the RWST.

Technical References: 2-AOP-RHR-1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-AOPRHR 3

Question Source: Bank # ______ IPEC Bank

Modified Bank # ______ Note changes or attach parent

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 5

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	0000152236	
		Equipment Control - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	
	Importance	3.1	

Question # 3

Given:

- Plant heatup is in progress in accordance with 2-POP-1.1, Plant Heatup from Cold Shutdown Conditions.
- Final preparations are in progress to enter MODE 3.
- 24 RCP is in operation.
- MCC 28 is de-energized for post maintenance testing of the normal supply breaker.
- The Reactor Trip and Bypass Breakers are tagged out for Power Cabinet fuse clip replacement.

Can the unit enter MODE 3?

A. Mode 3 cannot be entered. Tech Specs requires four RCS Loops OPERABLE to enter Mode 3 regardless of the status of the status of the Rod Control System.

B. Mode 3 cannot be entered. Tech Specs requires two RCS Loops OPERABLE and one in operation to enter Mode 3 when the Rod Control System is NOT capable of rod withdrawal.

C. Mode 3 can be entered. Tech Specs requires one RCS Loop OPERABLE as long as the Rod Control System is not capable of rod withdrawal to enter Mode 3.

D. Mode 3 can be entered. Tech Specs requires two loops (RHR and/or RCS) OPERABLE and one in operation to enter Mode 3 regardless of the status of the status of the Rod Control System.

Answer: B

Explanation/Justification:

A. Incorrect. Plausible because Tech Specs does require two RCS loops in operation when rod control is capable of rod withdrawal. Only one RCS loop must be in operation and another loop OPERABLE to enter mode 3.

B. Correct. With MCC 28 de-energized a second RCP cannot be started (power to bearing lift pumps). Based on these conditions only one loop is OPEABLE and entry into mode 3 is not allowed.

C. Incorrect. Plausible because the statement is true, only one loop is required to be in operation; however, a second loop must OPERABLE.

D. Incorrect. Plausible because the combination of 2 loops is the LCO for MODE 4.

Technical References: Technical Specifications

Proposed References to be provided: None

Learning Objective: I2LP-ILO-POP005 - 3

Question Source: Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 5

55.43 (b) 2

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000026K302	
		Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: - The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS	
	Importance	3.6	

Question # 4

The plant is in a normal full power lineup. During I&C troubleshooting, a technician inadvertently depressed Train B Containment Spray manual actuation pushbutton. What effect will this have on the Component Cooling Water System?

	CCW Pumps	**Aux CCW Pumps**	**ØB valves**	**CCW from RHR Hx**
A.	No Change	No Change	4 valves closed	No Change
B.	All Running	All Running	All Valves Closed	Both Valves Open
C.	All Running	All Running	4 valves closed	One Valve Open
D.	No Change	No Change	All Valves Closed	No Change

Answer: A

Explanation/Justification:

A single manual pushbutton for Containment Spray will actuate a single train of Spray (1 pump and half of the valves). It will also actuate a single train of ØB isolation. A manual containment spray actuation will not actuate a safety injection. An automatic spray actuation signal will actuate safety injection if not already actuated.

A. Correct. SI is not actuated and 4 of the 7 ØB valves will close (those controlled by Train B ØB).

B. Incorrect. Plausible because the candidate may believe that a single manual pushbutton will actuate both trains of ØB valves and initiate an automatic SI.

C. Incorrect. Plausible because the candidate may believe that a single manual pushbutton will actuate one trains of ØB valves and initiate an one train of SI.

D. Incorrect. Plausible because the candidate may believe that the manual pushbutton will actuate both trains of ØB valves.

Technical References: Logic Diag 225105-1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-ESS001 – 5

Question Source: Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 7

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000027A215	
		Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: - Actions to be taken if PZR pressure instrument fails high	
	Importance	3.7	

Question # 5

Given the following:

- Unit is operating at 100% power.
- A failure of the controlling pressurizer pressure channel caused actual pressurizer pressure to rise approximately 30 psig above normal.
- Pressurizer Pressure Master Controller was placed in MANUAL.

Which ONE of the following describes the action required to reduce RCS pressure?

A. Decrease the controller output.

B. Increase the controller output.

C. Lower the pressure setpoint adjustment.

D. Raise the pressure setpoint adjustment.

Answer: B

Explanation/Justification:

A. Incorrect and plausible an operator may think he has to lower the output to lower pressure
B. Correct
C. Incorrect but plausible because lowering the pressure setpoint would work in auto
D. Incorrect but plausible because of confusion between how the controller works in auto vs. manual

Technical References:	NA
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-RCSPZR - 9

Question Source:	Bank #	X	IPEC Bank4233
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000029K206 Knowledge of the interrelations between the ATWS and the following: - Breakers, relays, and disconnects	
	Importance	2.9	

Question # 6

Reactor Trip and Bypass Breakers have been aligned to support testing of Reactor Protection Train A when an inadvertent Safety Injection signal is generated on Safety Injection Train A. Which of the following describes Reactor Protection System response?

A. An ATWS occurs because Reactor Trip Breaker A and Reactor Trip Bypass Breaker B remain closed.

B. An ATWS occurs because Reactor Trip Breaker B and Reactor Trip Bypass Breaker A remain closed.

C. The Reactor trips. Reactor Trip Breaker A, Reactor Trip Breaker B, and Reactor Trip Bypass Breaker A are open. Reactor Trip Bypass Breaker B is racked out.

D. The Reactor trips. Reactor Trip Breaker A, Reactor Trip Breaker B, and Reactor Trip Bypass Breaker B are open. Reactor Trip Bypass Breaker A is racked out.

Answer: C

Explanation/Justification:

Meets KA 000029EK2.06 because the KA calls for knowledge of interrelations between trip breakers and ATWS. Since the question tests the knowledge of whether or not this breaker configuration can lead to an ATWS, the KA is met.

A. Incorrect but plausible because an operator may think that the SI Train A only causes a Reactor Trip on RPS Train A

B. Incorrect but plausible because an operator may make the same error as for A plus not recall how trip and bypass breakers are paired.

C. Correct

D. Incorrect but plausible if an operator does not recall how trip and bypass breakers are paired.

Technical References: System Description 16

Proposed References to be provided: None

Learning Objective: I2LP-ILO-ICROD - 9

Question Source: Bank # ______ IPEC Bank

Note changes or attach parent

Modified Bank # ______

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 7

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000038A201	
		Ability to determine and interpret the following as they apply to a SGTR: - When to isolate one or more S/Gs	
	Importance	4.1	

Question # 7

Given:

- Unit 2 has just experienced a steam generator tube rupture in 22 steam generator (SG).
- E-0, Reactor Trip or Safety Injection, actions have led to a transition to E-3, Steam Generator Tube Rupture.
- The operator has attempted to close 22 main steam isolation valve (MSIV). 22 MSIV failed to close.

What is the next action that must be taken to limit the release of radioactivity from 22 SG?

A. Continue attempts to close the 22 MSIV and cool down with all intact Steam Generators using Steam Generator Atmospheric Valves leaving remaining MSIVs open.

B. Continue attempts to close the 22 MSIV and transition to ECA-3.1, SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired.

C. Isolate the remaining Steam Generators by closing their MSIVs and cool down using intact Steam Generator Atmospheric Valves.

D. Isolate the remaining Steam Generators by closing their MSIVs and transition to ECA-3.1, SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because while not what is specified in E-3, the non-return check valves will actually make this work.
B. Incorrect Plausible because transition to ECA-3.1 would mitigate the event, but it is not what is specified in the EOP network.
C. Correct. This method isolates remaining SG from ruptured SG.
D. is plausible because transition to ECA-3.1 would mitigate the event, but it is not what is specified in the EOP network.

Technical References: 2-E-3

Proposed References to be provided: None

Learning Objective: I2LP-ILO-AOPSG1 - 2

Question Source: Bank # X IPEC Bank 8668

Modified Bank # ____ Note changes or attach parent

New ____

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ____

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000054A102	
		Ability to operate and/or monitor the following as they apply to the Loss of Main Feedwater (MFW): - Manual startup of electric and steam-driven AFW pumps	
	Importance	4.4	

Question # 8

Given the following plant conditions:

- Plant is at 50% power
- 21 Rod Drive MG set is out of service for bearing replacement
- A fault occurs on 3A 480V bus, coincident with a unit trip

Which ONE of the following statements describes the configuration of the auxiliary feedwater system, prior to any operator actions?

A. Only 22 and 23 AFW pumps running, AFW flow will be automatically established to 23 and 24 S/G's. Manual action will be required to establish feed to 21 and 22 SGs using 22 AFW pump.

B. Only 22 and 23 AFW pumps running, AFW flow will be automatically established to all S/G's.

C. Only 21 and 22 AFW pumps running, AFW flow will be automatically established to 21 and 22 S/G's. Manual action will be required to establish feed to 23 and 24 SGs using 22 AFW pump.

D. Only 22 AFW pump running, AFW flow will NOT be established to any SG. Manual action will be required to establish feed to all SGs using 22 AFW pump.

Answer: A

Explanation/Justification:

A. Correct because shrink will cause 23 and 22 to start. Manual action is needed to establish feed with 22 AFW pump. 50% initial power makes it not obvious as to whether or not sufficient shrink will occur. However, it does occur in the simulator and the answers do not provide for any other possibility.
B. Incorrect but plausible. 22 and 23 AFW pumps will start, but 22 will not feed SGs.
C. Incorrect and plausible. 21 AFW running is plausible since operator may confuse information in question or power supplies to pumps. 23 not running is plausible because of misunderstanding BO logic (note that U3 logic would support the 6A pump not running)
D. Incorrect and plausible. This is plausible because of potential confusion with the BO logic and not knowing if sufficient shrink would have occurred.

Technical References: System Description 27.1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-MFW001 - 5

Question Source: Bank # X IPEC Bank - 8710

Modified Bank # ______ Note changes or attach parent

New ______

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 7

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000055A201	
		Ability to determine and interpret the following as they apply to a Station Blackout: - Existing valve positioning on a loss of instrument air system	
	Importance	3.4	

Question # 9

A complete station blackout occurs. With no operator actions, how will the following plant equipment be affected?

A. No Charging Pumps Running

VC Monitors R-41/R-42 Supply/Return (PCV-1234, 1235, 1236, 1237) - closed

Diesel Generator Cooler Outlets (FCV-1176, 1176A) - open

B. No Charging Pumps Running

Main Feedwater Regulating valves (FCV-417, 427, 437, 447) - closed

Bypass Feedwater Regulators (FCV-417L, 427L, 437L, 447L) - open

C. Atmospheric Dump Valves (PCV-1134-1137) - open

CST to Hotwell Makeup (LCV-1128) - open

Non-Regenerative Heat Exchanger (TCV-130) - open

D. Pressurizer Spray valves (PCV-455A, 455B) - closed

Loop Charging (204A/204B) - closed

Charging Control (HCV-142) - closed

Answer: A

Explanation/Justification:

A. Correct
B. Incorrect but plausible. An operator may misunderstand the failure position of the bypass FRVs
C. Incorrect but plausible. An operator may misunderstand the failure positon of the ADVs.
D. Incorrect but plausible. An operator may misunderstand the failure position of the loop charging isolations.
Additionally, since trips/SI's without a complete station blackout will usually not lead to a loss of IA, an operator may pick a distractor for that reason as well.

Technical References: System Description 27.1, 2-AOP-AIR-1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EDS01 – 7
I2LP-ILO-EDS01 – 17

Question Source: Bank # IPEC Bank

Modified Bank # X Note changes or attach parent4200

New

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge:

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 7

55.43 (b)

Comments:

D. Correct. Saturation for 2050 psig is approximately 641 degrees F. Subcooling at this point is still 13 degrees but must be 19 degrees for Natural Circ verification

Technical References:		2-ES-0.2	
Proposed References to be provided:		ES-0.1 Attachment 3	
Learning Objective:		I2LP-ILO-EOPE00 - 1	
Question Source:	Bank #	Callaway 2007	IPEC Bank
	Modified Bank #	X	Note changes or attach parent Attached
	New		
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:	X	
10 CFR Part 55 Content:	55.41	(b) 10	
	55.43	(b)	
Comments:			

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000056K104	
		Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: - Definition of saturation conditions, implication for the systems	
	Importance	3.1	

Question # 10

Given the following conditions:

- The plant tripped due to a grid disturbance and loss of off-site power.
- The crew is performing actions of ES-0.1, Reactor Trip Response.
- RCS pressure is currently 2050 psig.
- CETs indicate 628 degrees F and increasing slowly.

Which ONE of the following describes the conditions currently present in the RCS, and the status of natural circulation flow in accordance with ES-0.1, Reactor Trip Response, Attachment 3 Natural Circulation Verification?

A. Saturated conditions; Natural Circulation flow in the RCS is established.

B. Subcooled conditions; Natural Circulation flow in the RCS is established.

C. Saturated conditions; Natural Circulation flow in the RCS is NOT established.

D. Subcooled conditions; Natural Circulation flow in the RCS is NOT established.

Answer: D

Explanation/Justification:

A. Incorrect. Not at saturation
B. Incorrect. Natural Circ is not verified because subcooling is too low. Must be 19 degrees F
C. Incorrect. Conditions do not indicate saturation

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000057A104	
		Ability to operate and/or monitor the following as they apply to the Loss of Vital AC Instrument Bus: - RWST and VCT valves	
	Importance	3.5	

Question # 11

The following plant conditions exist:

- Instrument Bus 21/21A has been lost due to inverter failure.
- Pressurizer Level Channel 1 is in defeat.
- Makeup Mode Selector switch is in AUTO

What is the impact in the CVCS system due to the loss of Instrument Bus 21/21A?

A. Letdown isolation will occur. Automatic makeup will not occur.

B. Valve 112B will open and 112C will close. Automatic makeup will not occur

C. Letdown isolation will not occur. Automatic makeup will not occur.

D. Valve 112B will open and 112C will close. Automatic makeup will occur.

Answer: D

Explanation/Justification:

A. Incorrect. Plausible because letdown isolation will not occur if PRZR Level Channel 1 is defeated
B. Plausible because Charging pump suction will shift to RWST, but Auto makeup will occur.
C. Incorrect. Plausible because Letdown Isolation will not occur, but Auto makeup will occur
D. Correct

Technical References:		2-AOP-IB-1 Attachment 1	
Proposed References to be provided:		None	
Learning Objective:		I2LP-ILO-AOPIB1 - 1	
Question Source:	Bank #		IPEC Bank
	Modified Bank #	X	Note changes or attach parent 16763
	New		
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:	X	
10 CFR Part 55 Content:	55.41	(b) 7	
	55.43	(b)	
Comments:			

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	0000582236	
		Equipment Control - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	
	Importance	3.1	

Question # 12

The following conditions exist at Unit 2:

- The unit is in MODE 2 preparing for a Reactor startup.
- Maintenance is performing troubleshooting on 21 Battery Charger due to log reading trends on charger output voltage.
- 21 Battery Charger trips and 21 DC Voltage is 108V.

Which one of the following actions is required?

A. Shut down 21 Static Inverter.

B. Transfer 21 Static Inverter to its Alternate Feed.

C. Cross-connect 21 and 22 DC Buses.

D. Open all reactor trip and reactor trip bypass breakers.

Answer: B

Explanation/Justification:

A. Incorrect but plausible. 21 Static Inverter is shutdown if unable to transfer to alternate feed
B. Correct
C. Incorrect but plausible. Cross connecting DC busses is allowed only in Mode 5.
D. Incorrect but plausible. An operator may believe this is necessary since the unit is in Mode 2.

Technical References: 2-AOP-DC-1

Proposed References to be provided:		None	
Learning Objective:		I2LP-ILO-AOPDC1 - 1	
Question Source:	Bank #	X	IPEC Bank 16628
	Modified Bank #		Note changes or attach parent
	New		
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:	X	
10 CFR Part 55 Content:	55.41	(b) 10	
	55.43	(b) 5	

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	0000622242	
		Equipment Control - Ability to recognize system parameters that are entry level conditions for Technical Specifications.	
	Importance	3.9	

Question # 13

Which of the following events are entry conditions for Technical Specifications assuming the plant is in Mode 1?

1. One Non-Essential Service Water Pump inoperable
2. One Essential Service Water Pump inoperable
3. TCV-1103 Containment Building Air Temp Control Valve fails closed
4. FCV-1176 Diesel Generator Cooling Water fails closed
5. NPO finds FCV 1111, SWP 24/25/26 SUP TO CONV NON ESSEN STOP and FCV 1112, SWP 21/22/23 SUP TO CONV NON ESSEN STOP open 50% each
6. Swapping from 1, 2, 3 Service Water Header to 4, 5, 6 Service Water Header as essential

A. 2, 4, 5

B. 1, 2, 6

C. 2, 4, 6

D. 2, 3, 4

Answer: A

Explanation/Justification:

1. Incorrect. TS requires 2 NESW pumps
2. Correct
3. Incorrect. TCV-1103 is not required (TCV-1104 and 1105 are required)
4. Correct
5. Correct Essential and Non-Essential headers are required to be isolated from each other
6. Incorrect. Note allows 8 hours to split headers.

A. Correct
B. Incorrect
C. Incorrect
D. Incorrect

Technical References: Tech Spec 3.7.8

Proposed References to be provided: None

Learning Objective: I2LP-ILO-SW001 - 10

Question Source: Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ____

10 CFR Part 55 Content: 55.41 (b) 5

55.43 (b) 2

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000077AK103	
		Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: - Under-excitation	
	Importance	3.3	

Question # 14

Given the following conditions:

- Plant is at 100% power 1070 MWe
- Generex is in AC Control
- Generator H2 Pressure is 60 psig

The System Operator has notified the plant that system grid voltage is high and forecasted to go higher.

If the System Operator requests the plant to take in the maximum value of MVARs to help stabilize the grid.

Using Graph EL-1, what is the maximum allowed MVAR incoming value, and how is the adjustment made?

	MAX INCOMING VALUE	METHOD OF ADJUSTMENT
A.	410 MVARs	AC Raise/Lower Switch
B.	410 MVARs	DC Raise/Lower Switch
C.	490 MVARs	AC Raise/Lower Switch
D.	490 MVARs	DC Raise/Lower Switch

Answer: A

Explanation/Justification:

A. Correct. Candidate must use the limit of the Under Excited Reactive Ampere Limit (URAL) to determine Maximum VARs IN versus the generator hydrogen pressure.

B. Incorrect. Plausible because the reactive load value is correct; however, the method of adjustment is incorrect with the GENEREX in AC Control. Adjustments using the DC Raise Lower Switch will be corrected to AC setpoint when in AC control.

C. Incorrect. Plausible because the reactive load value is incorrect but it is the value obtained if the candidate uses the hydrogen pressure curve instead of the URAL curve; the method of adjustment is correct with the GENEREX in AC Control.

D. Incorrect. Plausible because the reactive load value is incorrect but it is the value obtained if the candidate uses the hydrogen pressure curve instead of the URAL curve; the method of adjustment is incorrect with the GENEREX in AC Control.

Technical References:		Graph EL-1	
Proposed References to be provided:		Graph EL-1	
Learning Objective:		I2LP-ILO-MTG02 - 8 I2LP-ILO-MTG02 - 2	
Question Source:	Bank #	Watts Bar 2008	IPEC Bank
	Modified Bank #	X	Note changes or attach parent
	New		
Question History:	Last 2 NRC Exams at IPEC:		NA
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:		X
10 CFR Part 55 Content:	55.41	(b) 5	
	55.43	(b)	

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	00WE04K302	
		Knowledge of the reasons for the following responses as they apply to the LOCA Outside Containment: - Normal, abnormal and emergency operating procedures associated with LOCA Outside Containment	
	Importance	3.4	

Question # 15

Given the following:

- A LOCA outside containment has occurred.
- The crew is performing the actions in ECA-1.2, LOCA Outside Containment.

Which ONE (1) of the following actions will be attempted to isolate the break and which indication is used to determine if the leak has been isolated in accordance with ECA-1.2?

A. Isolate SI Hot Leg Injection piping; PZR level is monitored, because with the break isolated, RCS inventory will rapidly rise.

B. Isolate SI Hot Leg Injection piping; RCS pressure is monitored, because SI flow will repressurize the RCS with the break isolated.

C. Isolate RHR Cold Leg Injection piping; PZR level is monitored, because with the break isolated, RCS inventory will rapidly rise.

D. Isolate RHR Cold Leg Injection piping; RCS pressure is monitored, because SI flow will repressurize the RCS with the break isolated.

Answer: D

Explanation/Justification:

A. Incorrect. Plausible because candidate must remember that all Hot Leg Injection piping is inside VC. Also Inventory is not the condition monitored to determine if leak isolation is successful.

B. Incorrect. Plausible because candidate must remember that all Hot Leg Injection piping is inside VC. Also RCS pressure is the parameter monitored to determine if leak isolation was successful.

C. Incorrect. Plausible because RHR Cold Leg Injection piping is located outside the VC; however, Inventory is not the condition monitored to determine if leak isolation is successful.

D. Correct. Some RHR Cold Leg Injection piping is located outside the VC and RCS pressure is the parameter monitored to determine if leak isolation was successful.

Technical References: 2-ECA-1.2 BG

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPC12 - 3

Question Source: Bank # X IPEC Bank 16661

Modified Bank # Note changes or attach parent

New

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge:

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	00WE05K201	
		Knowledge of the interrelations between the Loss of Secondary Heat Sink and the following: - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	
	Importance	3.7	

Question # 16

Following a small-break LOCA and SI actuation, core exit TC's read 625°F and increasing.

- RCS pressure is 1400 psia and rising.
- S/G pressures are stable at 900 psig.
- Containment pressure is stable at 3 psig.
- The control room operators are attempting to establish MBFW flow in response to a loss of secondary heat sink.
- They are unable to lift the live lead on the feed water isolation relay signal.

Which one of the following describes the plant response?

A. The MBFW pumps cannot be reset to provide flow.

B. The SI signal cannot be reset.

C. The MBFW regulator valves cannot be opened from the control room.

D. Establishing MBFW flow will result in an excessively rapid RCS cooldown and depressurization.

Answer: C

Explanation/Justification:

A. Incorrect and plausible since an operator may confuse the function of lifting the relay lead belieiving it will affect MBFPs.
B. Incorrect but plausible since and operator may confuse function of lifting lead believing it affects SI reset.
C. Correct - with this lead in place, FRVs will not open due to the Reactor Trip with Low Tavg signal.
D. Incorrect but plausible because an operator may believe that failure to lift the lead could cause FRVs to be full open.

Technical References: 2-FR-H.1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPFH1 - 5

Question Source: Bank # X IPEC Bank 8170

Modified Bank # ____ Note changes or attach parent

New ____

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	00WE11A101	
		Ability to operate and/or monitor the following as they apply to the Loss of Emergency Coolant Recirculation: - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	
	Importance	3.9	

Question # 17

Given:

- A LOCA has occurred.
- Operators were performing ECA-1.1, Loss of Emergency Coolant Recirculation when Containment pressure is noted to be 24.3 psig.
- The decision of whether to remain in ECA-1.1 or transition to another procedure was properly made.

Which of the following describes how the Containment Spray system will be operated, and why?

The Containment Spray System is operated as directed in...

A. ECA-1.1 because it establishes minimum required containment spray flow and conserves RWST inventory.

B. FR-Z.1, Response to High Containment Pressure, since restoration of the critical safety function takes precedence.

C. ECA-1.1 since FRPs (Functional Restoration Procedures) are NOT implemented during the performance of ECA-1.1.

D. FR-Z.1 because containment is the last fission product barrier actions are the same as SAMGs (Severe Accident Management Guidelines)

Answer: A

Explanation/Justification:

A. Correct
B. Incorrect. Plausible because FR procedure typically have a higher priority than other emergency procedures. There are two procedures (ES-1.3 Transfer to Cold Leg Recirculation and ECA-0.0 Loss of All AC Power) that take priority over FRPs. The concept that a procedure may take prioity over the FRPs is not unrealistic. ECA-1.1 addresses a "best guess" approach to containment conditions where FR-Z.1 addresses a "worst case" approach. The need to conserver RWST inventory for core cooling is a higher priority
C. Incorrect. Plausible becauseECA-1.1 is a special case and candidates may believe that no FRPs are implemented. FRP are implemented in ECA-1.1 if the condition occurs with the exception of FR-Z.1 due to the need to conserve inventory for core cooling.
D. Incorrect. Plausible because containment is the final fission product barrier. In general actions in the SAMGs are focused on maintaining containment intact.

Technical References:	2-FR-Z.1
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-EOPC12 – 6

Question Source:	Bank #	X	IPEC Bank 24215
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 5
	55.43	(b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	00WE12K202	
		Knowledge of the interrelations between the Uncontrolled Depressurization of all Steam Generators and the following: - Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	
	Importance	3.6	

Question # 18

During the performance of ECA-2.1, Uncontrolled Depressurization of All Steam Generators, the following plant condition exists:

- Cooldown rate of the RCS is greater than 100°F/hr

How is the crew directed to control feedwater flow?

A. Feedwater flow is terminated to all but a single intact SG, which is fed at 85 gpm

B. Feedwater flow is reduced to 85 gpm to each SG with narrow range level less than 9%.

C. Feedwater flow is preferentially maximized to 22 or 23 SG until narrow range is > 10%

D. Total feedwater flow is maintained at 400 gpm until narrow range level in any SG is > 10%

Answer: B

Explanation/Justification:

A. Incorrect. Plausible because a similar strategy is used in FR-H.1 for feeding a hot dry SG.
B. Correct
C. Incorrect. Plausible because a similar strategy is used in several emergency procedures to ensure steam for the turbine driven AFW pump.
D. Incorrect. Plausible because 400 gpm is the minimum normal feedwater flow rate to ensure adequate heat sink if level is < 10%.

Technical References:	2-ECA-2.1
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-EOPC21 - 4

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 10
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000001A201	
		Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: - Reactor tripped breaker indicator	
	Importance	4.2	

Question # 19

Unit 2 is operating at 100% power

- Control Bank D rods start stepping out due to a Logic Cabinet Malfunction
- Operators manually trip the Reactor.
- 90 seconds after the initial pressing of the Reactor Trip Push Button, the button is pressed a second time during the read-through of E-0, Reactor Trip or Safety Injection.

After this action the following indications are observed:

- RTA – Green Light Lit
- RTB – No Lights Lit (bulbs and sockets are working correctly)

Based on these indications, which of the following is the next appropriate action to be taken by the team?

A. Manually insert control rods.

B. Dispatch NPO to locally trip the Reactor

C. Initiate Emergency Boration of the RCS

D. Verify Turbine Trip

Answer: D

Explanation/Justification:
This question requires the candidate to realize that 1)RTB did not open, but the reactor is tripped by RTB 2) That an automatic trip signal exists against RTB from SG low low level 90 seconds after trip from 100% power, and 3) That with this is the proper breaker indication for RTB in these circumstances.

A. Incorrect but plausible. Plausible because if the reactor was not tripped the next action would be to insert control rods per FR-S1.
B. Incorrect but plausible. Plausible because a candidate may believe this is done even if the reactor did trip or per FR-S1 if it did not.
C. Incorrect but plausible. Plausible because an operator may believe the reactor is not tripped and this action would be taken in FR-S1. Also plausible because an operator may believe this will be done in ES-0.1 due to breaker indications with a tripped reactor.
D. Correct. The reactor is tripped with one breaker open. The next procedure step is to verify turbine trip.

Technical References: System Description 16.1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-ICROD - 10

Question Source:

Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 6

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000032A201	
		Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: - Normal/abnormal power supply operation	
	Importance	2.8	

Question # 20

Unit 2 is in MODE 3 with the following conditions:

- Tave 547°F
- Pressurizer pressure 2235 psig
- Reactor trip breakers CLOSED
- Source range counts 52 cps (N31) and 55 cps (N32)
- Source range HV Manual On/Off is in NORMAL
- ALL Control Rod Banks are INSERTED
- ALL Shutdown Rod Banks are Withdrawn to 223 Steps

An I&C technician is troubleshooting power source problems with the NIS drawers that were noted a few days earlier following a reactor trip. During the troubleshooting activities, the following indications are received at the main control boards:

- SOURCE RANGE LOSS OF DETECTOR VOLT actuates.
- Source range counts: 52 cps (N31), 0 cps (N32)
- Reactor trip breakers CLOSED

Which ONE of the following describes what the I&C technician did?

A. Removed the CONTROL POWER fuses for N32 with the Level Trip switch in BYPASS

B. Removed the INSTRUMENT POWER fuses for N32 with the Level Trip switch in BYPASS.

C. Activated the RPS input for the SOURCE RANGE BLOCK.

D. Removed power simultaneously to TWO Power Range channels.

Answer: B

Explanation/Justification:

A. Incorrect. Removal of CONTROL POWER fuses will not result in loss of high voltage to the detector (supplied from the Instrument Power fuses) and thus would not result in loss of N32 indication. Regardless of Level Trip Bypass switch position, removal of the control power fuses would result in reactor trip from this condition.
B. Correct. INSTRUMENT POWER supplies the High Voltage for the detector. Control power supplies power to the protection bistables including the Level Trip Bypass circuit.
C. Incorrect. Plausible because RPS input for SR Block de-energizes High Voltage to both SR NIS channels. Indication that only one channel has lost indication should eliminate this distractor. Candidate must distinguish between conditions that result in de-energization of a single channel and both channels.
D. Incorrect. Plausible because removal of power to 2 PR NIS channels will result in removal of High Voltage to both SR NIS. This can be bypassed using the SR HV Manual On/Off switches. Candidate must recognize that the HV Manual On/Off switch is not in the correct position to allow de-energizing more than one power range instruments

Technical References: System Description 13.1
Proposed References to be provided: None

Learning Objective: I2LP-ILO-ICEXC - 7
I2LP-ILO-ICEXC - 9

Question Source:
Bank # DC Cook 2002 IPEC Bank
Modified Bank # X Note changes or attach parent
New

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level:
Memory or Fundamental Knowledge:
Comprehension or Analysis: X

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000033A103	
		Ability to operate and/or monitor the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: - Manual restoration of power	
	Importance	3	

Question # 21

Given the plant is at 100% power.
Intermediate Range Channel N-35 was removed from service due to a failure of the high voltage power supply.
I&C has completed repairs to Intermediate Range Channel N-35.

The major actions to return N-35 to service are listed below.

1. Install Instrument and Control Power Fuses (warm up for 30 minutes)
2. Verify Level Trip Switch is in BYPASS
3. Remove blocking strips for Reactor Trip and Rod Stop
4. Perform Bistable setpoint verification.
5. Place Level Trip Switch to NORMAL

Which of the following identifies the proper sequence for restoration of Intermediate Range Channel N-35?

A. 1, 2, 3, 4, 5

B. 2, 1, 5, 3, 4

C. 1, 3, 2, 5, 4

D. 2, 1, 4, 3, 5

Answer: D

Explanation/Justification:

Verify Level Trip Switch is in BYPASS. Should be first action to prevent possible trip
Install Instrument and Control Power Fuses (warm up for 30 minutes). Second to energize the drawer
Perform Post Maintenance Testing. After drawer is energized then ensure it functions properly
Remove blocking strips for Reactor Trip and Rod Stop. After drawer is verified to be functioning properly then remove blocking strips.
Place Level Trip Bypass Switch to Normal. The final step to return to service

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

Technical References:	2-SOP-13.1, attachment 4
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-ICEXC -11

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 10
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000036K101 Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents: - Radiation exposure hazards	
	Importance	3.5	

Question # 22

Given the following conditions:

- Irradiated fuel assemblies are being shuffled in the Spent Fuel Pool (SFP),
- An irradiated fuel assembly has been lifted clear of the racks and is in transit toward its new assigned position,
- SFP level is noted to be dropping slowly,
- The Fuel Transfer Canal Gate is closed and latched.

Which ONE of the following describes the preferred course of action in accordance with 2-AOP-FH-1 regarding the irradiated fuel assembly?

A. Place the assembly in an appropriate location.
B. Return the assembly to its original location in the racks.

C. Lower the assembly to the bottom of the SFP and check the gate seal inflated.

D. Continue moving the assembly toward the new location and check the gate seal inflated.

Answer: A

Explanation/Justification:

A. Correct. AOP-FH-1 directs this action
B. Incorrect. Plausible because candidates may be concerned with SFP Zone requirements.
C. Incorrect. Plausible because this action is similar to an action used inside the VC.
D. Incorrect. Plausible because candidates may be concerned with SFP Zone requirements.

Technical References:	2-AOP-FH-1		
Proposed References to be provided:	None		
Learning Objective:	I2LP-ILO-FHD001 - 11		
Question Source:	Bank #	X	IPEC Bank 23994
	Modified Bank #		Note changes or attach parent
	New		
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:	X	
10 CFR Part 55 Content:	55.41	(b) 7	
	55.43	(b)	

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	0000032431	
		Emergency Procedures/Plan - Knowledge of annunciators alarms, indications, or response instructions.	
	Importance	4.2	

Question # 23

The following conditions exist on Unit 2:

- Power is at 75% during a power ascension.
- Rods were being withdrawn to maintain Tavg on program.
- The C3 Rod Drive shaft disconnected from its spider hub.
- The Rod Drive shaft remains aligned with its bank.
- The Rod Control Cluster Assembly has fully inserted into the fuel assembly guide tubes.

Which of the following identifies the Alarms expected for this event?

	C3 Rod Bottom Light	Rod Bottom Rod Stop	NIS Power Range Dropped Rod Rod Stop	Rod Control Urgent Failure
A.	ON	ON	ON	OFF
B.	OFF	ON	OFF	ON
C.	ON	OFF	OFF	ON
D.	OFF	OFF	ON	OFF

Answer: D

Explanation/Justification:

C3 Rod Bottom Light comes from IRPI which actually measures drive shaft position. Since the drive shaft is still fully withdrawn, this light will be OFF

Rod Bottom Rod Stop comes from IRPI which actually measures drive shaft position. Since the drive shaft is still fully withdrawn, this alarm will be OFF

NIS Dropped Rod Rod Stop is generated from Power Range NIS decreasing at greater than 5% in 5seconds. This alarm should be ON

Rod Control Urgent Failure is not expected when the rod drops; however, this alarm will be on for most dropped rod recovery.

A. Incorrect
B. Incorrect
C. Incorrect
D. Correct

Technical References: 2-AOP-ROD-1, 2-ARP-FCF

Proposed References to be provided: None

Learning Objective: I2LP-ILO-AOPROD - 6

Question Source: Bank # Braidwood 2002 IPEC Bank

Modified Bank # X Note changes or attach parent

New

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge:

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 2

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	0000742225	
		Equipment Control - Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	
	Importance	3.2	

Question # 24

Which of the following describes a basis for LCO 3.5.2, 'ECCS OPERATING'?

A. Three of the four accumulators provide enough water to fully recover the core before significant clad melt following a LOCA.

B. The boron concentration in the RWST prevents a return to criticality event following a main steam line break.

C. Maximum hydrogen generation from zirconium water reaction is ≤0.17 times the hypothetical amount generated if all zirconium were to react.

D. Three ECCS trains are required to ensure that sufficient flow is available, assuming a single failure affecting any one train.

Answer: D

Explanation/Justification:

A. Incorrect. Plausible because three of four accumulator will PARTIALLY recover the core before significant clad melt following a LOCA.
B. Incorrect. Plausible because the candidate must recognize that RWST is a separate LCO and the boron concentration limits the potential for a post trip return to criticality and achieve significant power.
C. ncorrect. Plausible because the maximum hydrogen generation is part of the basis of this LCO; however the limit is 0.01 times the hypothetical amount
D. Correct

Technical References: Technical Specifications 3.5.2 Basis

Proposed References to be provided: None

Learning Objective: I2LP-ILO-SIS01 - 11

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:	X	
	Comprehension or Analysis:		
10 CFR Part 55 Content:	55.41	(b) 10	
	55.43	(b)	

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	00WE03K201	
		Knowledge of the interrelations between the LOCA Cooldown and Depressurization and the following: - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	
	Importance	3.6	

Question # 25

The following plant conditions exist while the team is responding to SI reduction to reduce RCS injection flow during the performance of ES-1.2, Post LOCA Cooldown and Depressurization, for a small break LOCA.

- One charging pump is running
- Both RHR pumps are secured
- # 22 SI Pump is running
- # 24 RCP is running
- Containment pressure is 1.2 psig
- RCS Hot Leg temperatures are 330°F and trending down
- RCS subcooling is 110°F and trending up slowly
- Pressurizer level is 38% and stable

The team is evaluating conditions to stop the remaining SI pump.

Which of the actions below should the team take first at this time?

A. Stop #22 SI Pump

B. Start one RHR pump

C. Depressurize the RCS to refill the Pressurizer

D. Manually operate SI pumps as necessary

Answer: B

Explanation/Justification:

A. Incorrect. Plausible because subcooling is a significant value; however, the required subcooling for these conditions is 209. Candidate should recognize that inadequate subcooling exists.

B. Correct. Starting an RHR pump in "injection mode" will allow securing 22 SIP without the required subcooling as long as hot leg temperature is < 345 degrees F. This temperature ensures saturation conditions in the RCS are below the shutoff head of RHR pumps.

C. Incorrect. Plausible because this action would be performed if Pressurizer level was < 28%. Candidate should recognize that adequate pressurizer level exists.

D. Incorrect. Plausible because this action is on the foldout page if conditions degrade and require SI reinitiation. Candidate should recognize that conditions are not degrading (Pressurizer level stable and subcooling trending up).

Technical References:	2-ES-1.2
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-EOPS12 - 1

Question Source:	Bank #	X	IPEC Bank 24344
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 10
	55.43	(b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	00WE06K202	
		Knowledge of the interrelations between the Degraded Core Cooling and the following: - Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	
	Importance	3.8	

Question # 26

The following plant conditions exist:

- Inadequate core cooling conditions exists
- Operators unable to re-establish high pressure SI flow
- All Core Exit Thermocouples indicate greater than 1200°F
- Unable to establish RCP restart criteria

The CRS directs you to start the RCPs (one at a time) until CETs indicate less than 1200°F. Is the CRS's direction to restart the RCPs correct for these plant conditions?

A. NO, RCP start without adequate support conditions will result in seal failures and greater loss of inventory.

B. NO, RCP start will result in phase separation causing a deeper uncovery of the core.

C. YES, RCP start should be done regardless of support conditions since a seal failure LOCA would aid in event mitigation.

D. YES, RCP start should be done regardless of support conditions to extend the time before core damage will occur.

Answer: D

Explanation/Justification:

A. Incorrect but plausible. A candidate would believe we should not start RCPs because there are situations where we do not start RCPs, but that is based on SG level.
B. Incorrect but plausible. Same explanation as A.
C. Incorrect but plausible. LOCA flow often aids in heat removal, but not in this case. FR-C.1 conditions cannot be met without a loss of inventory, so this LOCA is unlikely to help.
D. Correct per EOP background

Technical References: 2-FR-C.1 Background

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPC01 - 1

Question Source: Bank # X IPEC Bank - 8481

Modified Bank # ____ Note changes or attach parent

New ____

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 5

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	00WE08K301	
		Knowledge of the reasons for the following responses as they apply to the Pressurized Thermal Shock: - Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics	
	Importance	3.4	

Question # 27

15 minutes ago, the plant experienced a main steamline break, from 100% power. Because of difficulties in closing the MSIVs, 23 and 24 SGs have blown dry.

Current Plant status is as follows:

- E-2, Faulted SG Isolation, is in progress.
- RCS temperature is 290°F and decreasing.
- SI flow is still being supplied to the RCS.
- Total AFW flow is 800 gpm.
- All RCPs have been stopped due to loss of cooling water.
- RCS Pressure is 1000 psig and steady.
- Attachment 1 of E-0, Reactor Trip or Safety Injection, has been completed.

Which of the following is of greatest immediate concern?

A. A crack could propagate in the reactor vessel wall due to a pressurized thermal shock event.

B. Injection of ECCS accumulator nitrogen into the RCS is imminent, natural circulation cooling will be limited.

C. Controlled cooldown will be a challenge when 24 RCP is started due to 23 and 24 MSIVs being open.

D. The loss of thermal driving head in the dry SGs will reduce the amount of natural circulation flow, due to stagnant coolant loops.

Answer: A

Explanation/Justification:

A. Correct. Red Path conditions have just been met for PTS. With this amount of AFW flow and SI flow, the plant is causing a big challenge in this safety function.
B. Incorrect but plausible. Accumulator injection is imminent and N2 injection could hamper natural circulation flow, but this is not the greatest immediate concern.
C. Incorrect but plausible. Eventually (after soak) 24 RCP should be started if available. If 23 and 24 MSIVs are not shut this will pose a challenge, but it is not the greatest immediate concern. NOTE: 24 RCP ALWAYS STARTED! LST
D. Incorrect but plausible. This statement is correct, but cooling at this point in the event is not an issue since the main challenge is due to too much cooling.

Technical References:	FR-P.1 Status Tree
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-EOPFP1 - 3

Question Source:	Bank #	X	IPEC Bank 24216
			Note changes or
	Modified Bank #		attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 2
	55.43	(b)

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 Reactor Coolant Pump	
	003 K1.12	Knowledge of the physical connections and/or cause-effect relationships between the RCP and the following systems	
		CCW	
	Importance	3.0	

Question # **28**
The plant is at 100% power.

The following events occur:

- At 1530, the 22 RCP BEARING COOLANT LOW FLOW Alarm annunciates.
- Upper bearing temperature 176ºF and rising at 5ºF/minute.
- Lower bearing temperature 186ºF and rising at 5ºF/minute.

Seal injection flow has been maintained to the RCP.

Which ONE of the following describes the MAXIMUM time allowed before the crew must stop the 22 RCP?

A. 1530

B. 1532

C. 1533

D. 1535

Answer B

Explanation/Justification:

A. Incorrect. Plausible because the bearing temperatures are close to but below the trip setpoints.

B. Correct because the procedure (2-AOP-CCW-001 step 4.3 of Rev. 3) specifies tripping the RCP if CCW is lost for 2 minutes.

C. Incorrect. Plausible because at this point the lower bearing temperature will exceed 200F

D. Incorrect Plausible because at this point both bearing temperatures will exceed 200F.

Technical References: 2-AOP-CCW-1

Proposed References to be provided:

Learning Objective: I2LP-ILO-RCSRCP 10

Question Source: Bank # IPEC Bank 19162

Modified Bank # X Note changes or attach parent

New

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis:

10 CFR Part 55 Content: 55.41 (b) (4)

55.43

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004000K507	
		Knowledge of the operational implications of the following concepts as they apply to the CVCS: - Relationship between SUR and reactivity	
	Importance	2.8	

Question # 29

Given the following:

- Unit 2 has run for two months after completing a refueling outage when a unit trip occurs.
- 24 hours later, a Reactor Startup is in progress.
- Shutdown Banks have been withdrawn and some Rod Control system troubleshooting is in progress which is delaying pulling the Control Banks.
- The ATC notices that SR SUR is negative and count rate is rapidly decreasing.

Which of the following is the cause?

A. Swapping charging pumps to a pump that was in service 30 days ago.

B. Excessive check valve leakage during a safety injection pump surveillance test.

C. Placing a CVCS mixed bed demineralizer in service that was last used during the refueling outage.

D. Watch Chemist performs chemical addition to adjust Lithium concentration.

Answer: C

Explanation/Justification:

A. Incorrect but plausible because an operator may not remember the boron letdown curve for BOL

B. Incorrect but plausible since the SI pumps circulate RWST water which has sufficient boron concentration to affect RCS temp. The choice is incorrect because at NOP, SI pumps should not affect RCS parameters.
C. Correct reference AOP-UC-1 and system descriptions.
D. Incorrect but plausible. This chemical adjustment should have no effect on power, but a candidate could believe it is possible because lithium level is related to boron level.

Technical References: AOP-UC-1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-CVCS - 15

Question Source:

Bank #		IPEC Bank
Modified Bank #		Note changes or attach parent
New	X	

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level:

Memory or Fundamental Knowledge:

Comprehension or Analysis: X

10 CFR Part 55 Content:

55.41 (b) 1, 4

55.43 (b) 6

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005000A101 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: - Heatup/cooldown rates	
	Importance	3.5	

Question # 30

The RCS is at mid-loop with all S/G primary manways removed. Which ONE of the following conditions would result in the core becoming uncovered earliest if a total loss of RHR occurred 120 hours after shutdown? (Assume NO operator action taken)

A. Cold leg nozzle dams are installed and there have been no vent paths established.

B. Cold leg nozzle dams are installed and the pressurizer manway has been removed.

C. Hot leg nozzle dams are installed and there have been no vent paths established.

D. Hot leg nozzle dams are installed and the pressurizer manway has been removed.

Answer: C

Explanation/Justification:

RCS at mid-loop with all S/G primary manways removed and hot leg nozzle dams installed provides the least water inventory. Without vent paths established, a bubble will form in the reactor vessel head causing the core to become uncovered. C is the correct answer

A. Incorrect but plausible because an operator could confuse whether cold leg or hot leg is worst situation

B. Incorrect but plausible for same reasons as A and because an operator may think that having the vent path could cause more mass loss

C. Correct

D. Incorrect but plausible because an operator may think that having the vent path could cause more mass loss

Technical References:		System Description 1.0	
Proposed References to be provided:			
Learning Objective:		I2LP-ILO-RCS001 - 13	
Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:	X	
10 CFR Part 55 Content:	55.41	(b) 5	
	55.43	(b)	

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005000A404 Ability to manually operate and/or monitor in the control room Controls and indications for closed cooling water pumps	
	Importance	3.1	

Question # 31

Which ONE of the following describes how the Recirculation Pump motors are cooled during an SI with blackout?

A. The Recirculation Pumps are designed to run without CCW cooling for 24 hours

B. CCW is re-established prior to starting the Recirculation Pumps and the Aux CCW Pumps provide additional cooling.

C. The Recirculation Pumps have a backup city water cooling system that will provide city water to the Recirculation Pump coolers on low pressure

D. The Recirculation Pumps have a shaft driven cooling pump that circulate CCW through the Recirculation Pump bearing cooler

Answer: B

Explanation/Justification:

A. incorrect but plausible. Some ECCS pumps are designed to run without CCW cooling for similar periods of time.

B. correct

C. incorrect but plausible. RHR, HHSI and Charging Pumps have backup cooling provided from city water.

D. incorrect but plausible. HHSI pumps have this feature.

Technical References: System Description 4.1

Proposed References to be provided: None

Learning Objective:	I2LP-ILO-CCW001 14		
Question Source:	Bank #		
	Modified Bank #		Note changes or attach parent
	New	X	
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:	X	
	Comprehension or Analysis:		
10 CFR Part 55 Content:	55.41	(b) 4	
	55.43	(b)	
Comments:			

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006000A302	
		Ability to monitor operation of the Emergency Core Cooling System including: Pumps	
	Importance	4.1	

Question # **32**

The purpose of the SI Pump Suction Low Pressure alarm is to alert the operator of loss of net positive suction head to the...

A. RHR and HHSI pumps during injection phase of SI.

B. Recirc and RHR pumps during recirculation.

C. HHSI pumps during low head to high head recirculation.

D. HHSI pumps during injection phase of SI and recirculation.

Answer: C

Explanation/Justification:

This alarm is activated by a switch on the supervisory panel or when Recirc Switch 6 is taken to ON. The alarm is active when RHR or Recirc pumps are supplying flow to the suction of the SI Pumps (low head to high head recirculation or Hot Leg recirculation phase)

A. A. is incorrect but plausible. Candidate must recall which pumps and under which conditions the alarm is active

B. is incorrect but plausible. Candidate must recall which pumps and under which condition the alarm is active

C. Correct

D. incorrect but plausible Candidate must recall which pumps and under which condition the alarm is active

Technical References: ARP SBF-1

Proposed References to be provided: None

Learning Objective: __________

Question Source: Bank # __________ IPEC Bank

Modified Bank # X Note changes or attach parent (1929)

New __________

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: __________

10 CFR Part 55 Content: 55.41 (b) (7)

55.43 __________

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	006000K504 Knowledge of the operational implications of the following concepts as they apply to the ECCS: - Brittle fracture, including causes and preventative actions	
	Importance	2.9	

Question # **33**

FR-P.1, "Response to Imminent Pressurized Thermal Shock", contains SI termination criteria that will terminate SI when all conditions required for SI termination in other EOPs have not been satisfied.

Which one of the following is the reason for different criteria in FR-P.1?

A. Continued SI flow will add mass and could lead to challenging PZR Safety Valves on subsequent heatup.

B. SI flow may have contributed to the RCS cooldown.

C. RCS heat removal is via the steam generators and SI flow is NOT required.

D. The other SI termination criteria will have already been met when FR-P.1 is entered.

Answer: B

Explanation/Justification:

A. is incorrect and plausible. While added mass could add to re-pressurization effects, this is not the reason for SI termination. PZR Safety Valves are referred to so the choice is an incorrect statement.

B. is correct. FR-P1 background recovery/restoration technique section describes this as reason for early termination.

C. is incorrect and plausible. These conditions are often correct for entries into FR-P.1, but not always.

D. is incorrect and plausible. This statement is often true when FR-P.1 is entered but not always.

Technical References:	2-FR-P.1 Background
Proposed References to be provided:	None
Learning Objective:	

Question Source:	Bank #	X	IPEC Bank **19174**
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) (7)
	55.43	

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007000K101 Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following systems: - Containment system	
	Importance	2.9	

Question # 34

The Pressurizer Relief Tank (PRT) can be drained to:

A. Containment Sump and Reactor Cavity Sump

B. Containment Sump and Reactor Coolant Drain Tank (RCDT)

C. Waste Holdup Tank and Containment Sump

D. Waste Holdup Tank and Reactor Coolant Drain Tank (RCDT)

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because PRT can be drained to Containment Sump. It cannot be drained to Rx Cavity Sump
B. Incorrect. Plausible because PRT can be drained to Containment Sump and to suction of RCDT pumps. A check valve prevents draining to RCDT.
C. Correct
D. Incorrect. Plausible because PRT can be drained to CVCS HUT (via RCDT pumps) and to suction of RCDT pumps. A check valve prevents draining to RCDT.

Technical References:

Proposed References to be provided: None

Learning Objective: I2LP-ILO-RCSPZR - 6

Question Source:	Bank #	X	IPEC Bank 1694
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 3
	55.43	(b) 7

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	0100002123	
		Conduct of Operations - Ability to perform specific system and integrated plant procedures during all modes of plant operation.	
	Importance	4.3	

Question # 35

Unit 2 is cooling down and depressurizing the RCS at the start of a refueling outage. During the process of lowering pressure from NOP to 900 psig, the following actions are performed. Select the answer that puts these actions in the proper order as pressure is lowered per 2-POP-3.3 Plant Cooldown - Hot to Cold Shutdown.

1. Monitor pressure using OPS pressure indicators PI-413K, PI-433K, or PI-443K
2. PZR pressure control must be transferred to manual
3. Block low pressurizer pressure safety injection
4. Monitor pressure using RCS hot leg pressure recorders PT-402 or PT-403

A. 2, 3, 4, 1

B. 3, 2, 4, 1

C. 2, 3, 1, 4

D. 3, 2, 1, 4

Answer: B

Explanation/Justification:

The correct sequence for lowering pressure is start by lowering pressure in automatic, which is the preferred method. While manual could be done it does not have to be done until 1700 psi (low range of automatic control) and the procedure specifies 1750 psi. At about 1900 psi, low pressure SI must be blocked so it is the first action of the choices given. The next is transfer pressure control to manual at 1750 psi. When pressure is between 1500 and 1700 psi, the only means of monitoring pressure is using PT-402 or 403, so this is the next step in the sequence. The final item is to use the OPS pressure monitors.

A. Incorrect but plausible. It is plausible (and would work) that manual control would be used to lower pressure, but that is not what procedure specifies.
B. Correct per POP-3.3
C. See A, also it is plausible an operator could be confused on when each pressure indicator is used.
D. See C

Technical References: 2-POP-3.3

Proposed References to be provided: None

Learning Objective: I2LP-ILO-POP002 - 1

Question Source: Bank # ______ IPEC Bank

Note changes or attach parent

Modified Bank # ______

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008000A204	
		Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - PRMS alarm	
	Importance	3.3	

Question # 36

The plant is operating at 100% Power.

ALL of the following have occurred:

Radiation Monitor R-47 is in alarm
RCV-017, CCW Surge Tank Vent Valve has automatically closed.
CCW Surge Tank High Level Alarm has annunciated.

Which of the following events could have caused these conditions to occur and what is the appropriate procedure to address the condition?

A. Large tube leak in RCP Seal Return HXs
Go to 2 AOP-RCP-1, Reactor Coolant Pump Malfunctions

B. Large tube leak in RCP Seal Return HXs
Go to 2-AOP-LICCW-1 Leakage into Component Cooling System

C. Large tube leak in Non-regenerative HX
Go to 2-AOP-CVCS-1, CVCS Malfunctions

D. Large tube leak in Non-regenerative HX
Go to 2-AOP-LICCW-1 Leakage into Component Cooling System

Answer: D

Explanation/Justification:

The only auto closure signal for RCV-017 is high activity on Radiation Monitor R-47.

A. Incorrect. Plausible because an operator may not recall the pressure differences involved. Seal return is from RCS, but it is essentially VCT pressure at this point. In addition AOP-RCP-1 is not the correct procedure to address this condition.

B. Incorrect. Plausible because an operator may not recall the pressure differences involved. Seal return is from RCS, but it is essentially VCT pressure at this point. In addition AOP-RCP-1 is not the correct procedure to address this condition.

C. Incorrect. Plausible because a large leak in the Non-regenerative heat exchanger will cause leakage into CCW and may cause RCV-017 to auto close. In addition AOP-CVCS-1 is not the correct procedure to address this condition.

D. Correct. A large leak in the Non-regenerative heat exchanger will cause leakage into CCW and may cause RCV-017 to auto close. AOP-LICCW-1 is the correct procedure to address this condition.

Technical References: System Description 4.1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-CCW001 – 9

Question Source: Bank # ______ IPEC Bank

Modified Bank # X Note changes or attach parent 18782

New ______

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 4

55.43 (b) 7

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010000K603	
		Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: - PZR sprays and heaters	
	Importance	3.2	

Question # 37

Offsite power is lost without SI actuation.

- The control room operators verify a reactor trip and a turbine trip.
- They determine that the EDG's have energized the 480 V AC busses.
- All appropriate loads have sequenced on.
- While ensuring that the RCS stabilizes at no-load conditions, an operator observes that PZR pressure is 2100 psig and slowly lowering.
- PZR PORV's and spray valves are closed.
- PZR level has risen from 19% to 25%.

What corrective action, if any, should be taken?

A. No operator action is necessary.

B. Manually actuate SI.

C. Maximize charging flow.

D. Reset the PZR Backup heaters.

Answer: D

Explanation/Justification:

A. Incorrect but plausible because an operator may not know when pressure will start to recover and think heaters are available

B. Incorrect but plausible because an operator may interpret data as indicating SI needed due to pressure

C. Incorrect but plausible since an operator may assume that heaters are not reset when buses are powered from EDGs this early in the response and that charging (pressurizer level) is used to raise pressure

D. Correct. Loss of off-site power blocks auto closure of backup heater breakers. This must be reset to allow breakers to close.

Technical References:	2-E-0 Reactor Trip or Safety Injection		
Proposed References to be provided:	None		
Learning Objective:	I2LP-ILO-EOPE00 - 4		
Question Source:	Bank #	X	IPEC Bank 3683
	Modified Bank #		Note changes or attach parent
	New		
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:	X	
10 CFR Part 55 Content:	55.41	(b) 5	
	55.43	(b)	

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012000K502	
		Knowledge of the operational implications of the following concepts as they apply to the RPS: - Power density	
	Importance	3.1	

Question # 38

The plant is at 80% power during a power ascension.

Assuming that RCS and flux distribution parameters remain on program/target, as power is raised 80% to 100%, how will the over-temperature (OT) and over-power (OP) differential temperature (DT) Reactor Protection setpoints change?

	OTΔT setpoint	OPΔT setpoint
A.	increase	stay the same
B.	stay the same	decrease
C.	decrease	stay the same
D.	stay the same	increase

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because the OPDT setpoint does remain the same, and the applicant may confuse the fact that the OTDT setpoint decreases with the term 'increase' meaning the actual value is closer to setpoint.

B. Incorrect. Plausible because the OPDT setpoint never increases from its nominal value. It will, however decrease if T-avg deviates above its nominal 100% power program value. This is not the condition described by the question. OTDT setpoint does change based on margin to DNB. As conditions change that place the reactor closer to DNB, the OTDT setpoint will decrease and vice versa.

C. Correct. Since T-avg at 80% power is less than 100% power, the OPDT setpoint will be at its nominal full power value and thus, will not change from 80 to 100% power assuming T-avg stays on program. The OTDT setpoint, on the other hand can increase or decrease from its nominal value. Since program T-avg will increase approximately 5 more degrees, the trip setpoint will become more limiting, decreasing to its nominal full power value.

D. Incorrect. OTDT decreases as referenced above in C. The OPDT setpoint never increases from its nominal value. It will, however decrease if T-avg deviates above its nominal 100% power program value

Technical References: Technical Specifications 3.3.1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-ICRXP - 3

Question Source:	Bank #	Wolf Creek 2009	IPEC Bank
	Modified Bank #		Note changes or attach parent
	New		

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level:
Memory or Fundamental Knowledge:
Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 1

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013000K413	
		Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: - MFW isolation/reset	
	Importance	3.7	

Question # 39

Which ONE of the following will result in a Main Feedwater Isolation with the plant initially operating at 75% power?

A. Pressure transmitter 419A for SG 21 has failed high and Pressure Transmitter 449A for SG 24 has failed HIGH.

B. Level transmitter for 437A for SG 23 has failed high and Level transmitter 447C for SG 24 has failed HIGH.

C. Containment pressure transmitter 948A has failed HIGH and Containment pressure transmitter 949B has failed HIGH.

D. Pressurizer pressure transmitter 456 (Channel 2) has failed LOW and Pressurizer pressure transmitter 474 (Channel 4) has failed LOW.

Answer: D

Explanation/Justification:

A. Incorrect. Plausible because condition requires evaluation to see if delta P SI signal will be generated. An SI signal will not be generated.
B. Incorrect. Plausible because High SG level causes feedwater isolation however the coincidence is not correct.
C. Incorrect. Plausible because Containment Pressure will cause a safety injection. The coincidence is 2 of 3 on the 948A-C instruments. The 949A-C transmitters are used for Containment Spray.
D. Correct. Pressurizer Low Pressure Reactor trip with low Tavg

Technical References: System Description 21

Proposed References to be provided: None

Learning Objective: I2LP-ILO-ESS001 – 5

Question Source:	Bank #		IPEC Bank
	Modified Bank #	X	Note changes or attach parent 4232
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013000K601	
		Knowledge of the effect of a loss or malfunction of the following will have on the ESFAS: - Sensors and detectors	
	Importance	2.7	

Question # 40

Given the following:

- The plant is at 100% power.
- All control systems are in their normal alignments.
- Pressurizer Pressure Transmitter PT-455 has failed LOW.
- All actions have been taken to remove the transmitter from service in accordance with 2-AOP-INST-1, Instrument or Controller Failures.

Which ONE of the following describes the logic required from the remaining operable pressurizer pressure channels to initiate (1) a Low Pressurizer Pressure Reactor Trip, and (2) a Low Pressurizer Pressure Safety Injection actuation?

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

B. (1) 1 out of 3
(2) 1 out of 2

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

D. (1) 1 out of 3
(2) 1 out of 3

Answer: B

Explanation/Justification:
Pressurizer Low Pressure Reactor Trip is a 2 out of 4 logic.
Pressurizer Low Pressure SI is a 2 out of 3 logic

A. Incorrect. Opposite of actual

B. Correct
C. Incorrect. Reactor Trip receives inputs from 4 channels
D. Incorrect. Safety Injection receives input from 3 channels

Technical References:	Logic Diag 225102		
Proposed References to be provided:	None		
Learning Objective:	I2LP-ILO-ESS001 – 4		
	I2LP-ILO-ICRXP - 3		
Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:	X	
	Comprehension or Analysis:		
10 CFR Part 55 Content:	55.41	(b) 5	
	55.43	(b)	
Comments:			

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022000K301	
		Knowledge of the effect that a loss or malfunction of the CCS will have on the following: - Containment equipment subject to damage by high or low temperature, humidity, and pressure	
	Importance	2.9	

Question # 41

Given:

- A small break LOCA occurred approximately 45 minutes ago.
- Containment pressure peaked at 8 psig.
- Containment pressure subsequently decreases to 2.5 psig.
- Containment radiation peaked at 15 R/hr on R-25 and 26
- The crew is performing an SI flow reduction using ES-1.2, Post-LOCA Cooldown and Depressurization.
- One SI pump has been secured using Adverse Containment Values.
- A second pump cannot be secured at this time.
- Subcooling remains constant.

Which of the following is true regarding securing subsequent SI Pumps?

A. The pump cannot be secured using normal values.
Once flow reduction is started using adverse containment values, subsequent flow reduction actions continue using adverse containment values.

B. The pump cannot be secured using normal values.
Since pump was evaluated using adverse containment values; it must be secured using adverse containment values. Subsequent actions continue using normal values.

C. The pump can be secured using normal containment values.
When adverse containment pressure and radiation values return to normal, normal containment values are used.

D. The pump can be secured using normal containment values. When containment pressure decreases to acceptable range, normal containment values are used.

Answer: D

Explanation/Justification:
Two conditions result in adverse containment conditions, VC pressure and radiation. If pressure decreases to less than the adverse value, the crew should use normal values. If radiation levels decrease to normal values, the crew should continue using adverse containment values.

A. Incorrect. Plausible because as discussed above, one conditions result in returning to normal values (pressure) and one condition (radiation) does not. It is reasonable that once the flow reduction using adverse values is initiated, it should continue using adverse values.
B. Incorrect. Plausible because as discussed above, one conditions result in returning to normal values (pressure) and one condition (radiation) does not. It is reasonable that once the pump itself was evaluated using adverse values is initiated; it should continue using adverse values.
C. Incorrect. If containment radiation levels return to normal range, adverse values are used until engineering evaluation allows use of normal values.
D. Correct

Technical References: OAP-12

Proposed References to be provided: NA

Learning Objective: I2LP-ILO-EOPROU – 21

Question Source: Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026000K201	
		Knowledge of bus power supplies to the following: - Containment cooling fans	
	Importance	3.0	

Question # 42

The following conditions exist:

- Reactor Trip and Safety Injection have actuated.
- Main Steam Line Isolation and Containment Isolation Phase B have actuated.
- 480 volt vital bus 5A is deenergized due to a fault.

Which of the following describes the equipment available to reduce containment pressure?

A. 22 CS pump, 23, 24, 25 Containment Fan Cooler Units

B. 21 CS pump, 21, 24, 25 Containment Fan Cooler Units

C. 21 CS pump, 22, 24, 25 Containment Fan Cooler Units

D. 22 CS pumps, 21, 24, 25 Containment Fan Cooler Units

Answer: A

Explanation/Justification:
IPEC has 3 Safeguards buses (5A, 2A-3A, and 6A) Any 2 safeguards buses satisfy minimum safeguard power requirements. Equipment is distributed among the safeguards buses. Candidates must know the what equipment is power from what safeguards bus. Bus 5A is the ony 480 V bus with 2 FCUs (Buses 2A and 3A each have 1 FCU)

A. Correct
B. Incorrect. Plausible because the 3 FCUs are correct. The power supply to 21 CS pump is 5A.
C. Incorrect. 21 CS Pump and 22 FCU are power from 5A.
D. Incorrect. 22 CS Pump and 22 FCU are power from 5A

Technical References: 2-AOP-480V-1 Att 5

Proposed References to be provided:		None	
Learning Objective:		I2LP-ILO-ESS001 – 3	
Question Source:	Bank #	X	IPEC Bank 17513
	Modified Bank #		Note changes or attach parent
	New		
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:	X	
	Comprehension or Analysis:		
10 CFR Part 55 Content:	55.41	(b) 7	
	55.43	(b)	
Comments:			

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039000K405	
		Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: - Automatic isolation of steam line	
	Importance	3.7	

Question # 43

A fault occurred on 23 SG inside containment, and a high steam flow was sensed only on 23 Main Steam Line. 23 SG pressure is 500 psig and decreasing. Which ONE of the following is correct?

A. All Main Steam Isolation Valves will close immediately.

B. Only the Main Steam Isolation Valve for the 23 Steam Generator will close.

C. All Main Steam Isolation Valves will close when 23 Steam Generator Pressure is 155 psid below the pressure in the other Steam Generators.

D. None of the Main Steam Isolation Valves will be immediately affected.

Answer: D

Explanation/Justification:

A. High flow conditions are required in two of the four steam generators concurrent with low Tave or low Pressure
B. Individual MSIVs close when the control switch is placed in TRIP or air is lost to that MSIV for an extended period of time
C. Steam Line differential pressure causes a safety injection signal not a MSIV isolation signal
D. Correct answer. Drawing 241685

Technical References: Drawing 241685

Proposed References to be provided: None

Learning Objective: I2LP-ILO-ESS001 - 5

Question Source:	Bank #	X	IPEC Bank 5149
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059000K302	
		Knowledge of the effect that a loss or malfunction of the MFW System will have on the following: - AFW System	
	Importance	3.6	

Question # 44

The plant was operating at 30% power during a power ascension with 21 and 22 MBFP operating in AUTO when 21 MBFP trips on low auto stop oil pressure.

When can 21 and 23 AFW pumps be secured and not auto-restart with the switches in auto?

A. After 21 MBFP Reset Switch is placed in trip.

B. Any time adequate MFW flow exists.

C. Any time SG level is greater than 9% in all SGs.

D. When 21 MBFP auto stop oil pressure returns to > 25 psig.

Answer: A

Explanation/Justification:

A. Correct (2-AOP-FW-1 Step 102)
B. Incorrect. Plausible because adequate feed flow would maintain SG level above the auto start setpoint
C. Incorrect. Plausible because 9% is the auto start setpoint; however, SG level will not decrease to 9% before the AFW pumps auto start on MBFP trip.
D. Incorrect. Plausible because 28 psig is the auto stop oil setpoint that will cause the MBFP to trip, so the pump trip cannot be reset at 25 psig.

Technical References: 2-AOP-FW-1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-MFW001 - 9

Question Source:	Bank #	IPEC Bank
	Modified Bank #	Note changes or attach parent
	New	X
Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X
10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b)
Comments:		

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061000A101	
		Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW System controls including: - S/G level	
	Importance	3.9	

Question # 45

Unit 2 was operating at 100% power when 22 RCP tripped due to a fault.

All equipment operated as designed.

Which of the following describes how 21 and 22 SG levels will respond to this event and why?

A. 22 SG level will increase at a faster rate than the 21 SG. AFW flow is greater to 22 SG because it is at a lower pressure

B. 22 SG level will increase at a slower rate than the 21 SG. AFW flow is lower to 22 SG because it is at higher pressure

C. 22 SG level will increase at a slower rate than the 21 SG because it is steaming at a higher rate.

D. 22 SG level will increase at a faster rate than the 21 SG because it is steaming at a lower rate.

Answer: D

Explanation/Justification:

A. Incorrect: Plausible because 22 SG level will increase at a faster rate, but AFW flow is automatically controlled at approximately 200 gpm to each SG

B. Incorrect: Plausible because candidate may confuse SG pressure response on trip of an RCP.

C. Incorrect: Plausible because 22 SG is one of the supplies to the TDAFW pump. Candidates may believe that this will cause the level increase to be slower.

D. Correct: With non-return check valves in the steam lines, 22 SG will steam at a much lower rate even though it is supplying 22 AFW pump turbine.

Technical References: ______

Proposed References to be provided: None

Learning Objective: I2LP-ILO-RCSRCP - 15

Question Source: Bank # ______ IPEC Bank

Modified Bank # ______ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 14

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062000A101	
		Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the A.C. Distribution System controls including: - Significance of D/G load limits	
	Importance	3.4	

Question # 46

While transferring the plant to Cold Leg Recirculation, it is noted that #21 EDG loading is currently 1650 KW. The next step to be performed in ES-1.3, Transfer to Cold Leg Recirculation, is to place Safety Injection Recirculation Switch 4 to "ON", which would start 21 Recirculation Pump (299 KW).

How does ES-1.3 address EDG loading at this step?

A. Load must be removed from the bus prior to starting the pump to prevent exceeding the maximum short time (2 hr) load rating of the EDG.

B. Load must be removed from the bus prior to starting the pump to prevent exceeding the continuous EDG load rating.

C. Starting this pump is allowed, but EDG load will be limited to the short time (2 hr) load limit after this pump is placed in service.

D. 22 Recirculation Pump will be manually started instead of 21 to prevent overloading 21 EDG.

Answer: C

Explanation/Justification:

This question requires the candidate to have the following knowledge:
1. what the continuous and 2 hour limits on EDGs are
2. Have a rough idea of KW of recirc pumps. Note that the KW values picked would be valid for any pump that would be run in ES-1.3. All safeguards motors are >100 KW and < 450 KW. Recirc pumps are 299 KW.
3. Procedure actions of ES-1.3 (we do not formally address EDG load).
Additionally, the candidate has to piece together the information to come to conclusion that starting this pump will put us over the normal limit, but within the short term limit and that this is OK.

A. Incorrect but plausible. It is possible that a candidate may believe that the short term limit would be exceeded.
B. Incorrect but plausible. The continuous load rating will be exceeded, but we do not have to shed load.
C. Correct. Per 2-sop-27.3.1.1 caution at step 4.2.10, this load is allowed for 2 hours in a 24 hour period.
D. Incorrect but plausible. It is reasonable for a candidate to assume that 22 pump would be used, but the procedure does not have steps to check for load prior to stating 21 pump with recirc switch 4.

Technical References: 2-SOP-27.3.1.1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EDSEDG - 8

Question Source: Bank # X IPEC Bank 3035

Modified Bank # ____ Note changes or attach parent

New ____

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 7

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062000A201	
		Ability to (a) predict the impacts of the following malfunctions or operations on the A.C. Distribution System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Types of loads that, if de-energized, would degrade or hinder plant operation	
	Importance	3.4	

Question # 47

The plant is at power with the following conditions present:

- 100% power
- All control systems are in automatic
- Tave – 565 degrees F
- RCS press – 2235 psig
- No equipment out of service
- Pressure channel 3 is in control
- Pressure channel 2 is the alarm channel

Which one of the following correctly describes one effect of losing 22 instrument bus?

A. PORV 456 is prevented from automatically opening. 2-AOP-IB-1, Loss of Power to an Instrument Bus, directs closing the block valve and removing power within 4 hours.

B. PORV 456 is prevented from automatically opening. 2-AOP-IB-1, Loss of Power to an Instrument Bus will restore automatic control by defeating the affected channel.

C. PORV 455C will not automatically open. 2-AOP-IB-1, Loss of Power to an Instrument Bus, directs closing the block valve and removing power within 4 hours.

D. PORV 455C is prevented from automatically opening. 2-AOP-IB-1, Loss of Power to an Instrument Bus will restore automatic control by defeating the affected channel.

Answer: B

Explanation/Justification:

A. Incorrect. Candidate must recall that PT-456 (Channel 2) is powered from 22 IB and loss of PT-456 will prevent PORV 456 from opening. PORV can be manually operated thus the TS action to place in close and remove power is in correct.

B. Correct. Candidate must recall that PT-456 (Channel 2) is powered from 22 IB and loss of PT-456 will prevent PORV 456 from opening. 2-AOP-IB-1 will defeat the affected channel and restore automatic control.

C. Incorrect. Plausible because the candidate must recall that PT-456 (Channel 2) is powered from 22 IB and that it cannot be in control. Thus it cannot affect PORV 455C. The Tech Spec action for an inoperable PORV is to place the block valve in close and remove power within 4 hours

D. Incorrect. Plausible because the candidate must recall that PT-456 (Channel 2) is powered from 22 IB and that it cannot be in control. Thus it cannot affect PORV 455C. 2-AOP-IB-1 will defeat affected channel, but channel 2 is not an input to PCV-455C.

Technical References: System Description 1.0

Proposed References to be provided: None

Learning Objective: I2LP-ILO-RCSPZR – 9

Question Source: Bank # ______ IPEC Bank

Modified Bank # ______ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

	Comprehension or Analysis:	X
10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063000A403	
		Ability to manually operate and/or monitor in the control room: - Battery discharge rate	
	Importance	3.0	

Question # 48

A 345KV fault leads to a trip of Unit 2 from 100% power.

Due to the electrical transient, 21 Battery Charger trips and cannot be restarted.

Based on these conditions and equipment design criteria, what is the expected plant response?

A. 21 Battery is designed to ensure voltage will remain above a predetermined acceptable value for 2 hours. After voltage drops below this level, control power to 480V switchgear and safeguards relays powered from 21 DC will switch to an alternate source.

B. 21 Battery is designed to ensure voltage will remain above a predetermined acceptable value for 2 hours. After voltage drops below this level, control power to 480V switchgear powered from 21 DC will switch to an alternate source.

C. 21 Battery is designed to ensure voltage will remain above a predetermined acceptable value for 4 hours. After voltage drops below this level, control power to 480V switchgear and safeguards relays powered from 21 DC will switch to an alternate source.

D. 21 Battery is designed to ensure voltage will remain above a predetermined acceptable value for 4 hours. After voltage drops below this level, control power to 480V switchgear powered from 21 DC will switch to an alternate source.

Answer: B

Explanation/Justification:

This question tests two knowledge areas. One being how long batteries are rated for (2 hours per T.S. Basis). The other being what automatically swaps to alternate power on a loss of DC, which is only switchgear.

A. Incorrect but plausible. The safeguards relays do not switch power source, but it a plausible answer because switchgear does.

B. Correct

C. Incorrect but plausible. There are many 4 hour ratings. It is plausible that a candidate could think 4 hours is correct. Also see above for including safeguards relays.

D. Incorrect but plausible. See above for 4 hours vs. 2 hours.

Technical References: Tech Spec 3.8.4 Basis

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EDS03 – 12

Question Source: Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ____

10 CFR Part 55 Content: 55.41 (b) 7

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064000K102	
		Knowledge of the physical connections and/or cause-effect relationships between the ED/G System and the following systems: - ED/G cooling water system	
	Importance	3.1	

Question # 49

What would happen if the Jacket Water Pump on a Emergency Diesel Generator had a broken shaft and the Emergency Diesel Generator received an AUTO start signal, with no operator action.

A. The Emergency Diesel Generator would run until it overheated, then high oil temperature would trip the 86 device

B. The Emergency Diesel Generator would start and continue to run, but the field would not 'flash' so there would be no generator output

C. The Emergency Diesel Generator would start but only run for about 13 seconds, then the 86 would trip

D. Without jacket water pressure the Emergency Diesel Generator would start, run for 2 minutes and shut down normally

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because the EDG would overheat without cooling. The engine start failure would trip the diesel before this occurrs.
B. Incorrect. Plausible because the EDG would overheat without cooling. It is also plausible since Jacket Water pressure is how the circuit determines that the engine came up to speed. It is reasonable that a candidate would believe the field flash depends on this. The engine start failure would trip the diesel before this occurs.
C. Correct

D. Incorrect. Plausible because the engine would start; however it would not run for 2 minutes nor would it shutdown normally.

Technical References:	2-ARP-003
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-EDSEDG – 10

Question Source:	Bank #	X	IPEC Bank 6756
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073000K301	
		Knowledge of the effect that a loss or malfunction of the PRM System will have on the following: - Radioactive effluent releases	
	Importance	3.6	

Question # 50

A liquid release is in progress. Power is lost to R-54, Liquid Radiation Monitor.

Assuming all components functioned as designed, what is the status of the Waste Distillate System?

	Waste Dist Trans Pump	WDTP Disch Valve	WDTP Recirc Valve	WDTP Common Disch Valve
A.	Tripped	Open	Closed	Closed
B.	Running	Closed	Open	Open
C.	Tripped	Closed	Open	Closed
D.	Running	Open	Closed	Open

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because the common discharge valve (LWR-701) will close and the pump trip which would stop the leak; however, the pump discharge valve will close and the recirc valve will open.

B. Incorrect. Plausible because closing the pump discharge valve and opening the pump recirculation valve will stop the release and continue mixing of the Distillate Storage Tank contents.

C. Correct

D. Incorrect. Candidate may not recognize that a loss of power to the radiation monitor results in the same automatic actions as a high radiation level alarm.

Technical References:	2-SOP-12.3.3
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-RMS001 - 2

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 13
	55.43	(b) 4

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076000A201	
		Ability to (a) predict the impacts of the following malfunctions or operations on the SWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Loss of SWS	
	Importance	3.5	

Question # 51

Given the following conditions:

- The plant is at 26% power during a plant startup
- Service Water is in three header operation
- #11 River Water pump is supplying Conventional loads

Subsequently, 11 River Water pump trips. The team enters 2-AOP-SW-1, (Service Water System Malfunction) and determines NEITHER River Water pump can be started.

For the above conditions, which one of the following actions is required by 2-AOP-SW-1?

A. Shift Service Water System operation to 2 header operation

B. Trip the reactor and go to E-0, (Reactor Trip or Safety Injection)

C. Commence a plant shutdown in accordance with POP-3.1, (Plant Shutdown, Mode 1 to Mode 3)

D. Trip the Turbine and go to 2-AOP-TURB-1, (Main Turbine Trip Without a Reactor Trip)

Answer: B

Explanation/Justification:

A. Incorrect but plausible because switching to two header operations would restore service water to all loads, however there will not be sufficient time to do this.

B. Correct answer based on Step 4.7 of 2-AOP-SW-1.

C. Incorrect but plausible. If the loss of the River Water Pumps did not affect the conventional header, a shutdown per the POP would be called for (step 4.45 of 2-AOP-SW-1)

D. Incorrect but plausible because if power was below P-8 this would be the correct action.

Technical References: 2-AOP-SW-1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-AOPSW1-3

Question Source:

Bank #	X	IPEC Bank 8713	
Modified Bank #		Note changes or attach parent	
New			

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level:

Memory or Fundamental Knowledge:	
Comprehension or Analysis:	X

10 CFR Part 55 Content:

55.41	(b) 4
55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	0760002132	
		Conduct of Operations - Ability to explain and apply all system limits and precautions.	
	Importance	3.8	

Question # 52

Which ONE of the following describes the starting limitations as per 2-SOP-24.1 for the Service Water Pumps? Assume pump started at ambient conditions and coasts to a stop between starts.

A. Two consecutive starts are allowed, a third start is allowed after pump has been idle for a minimum of 30 minutes.

B. After the first start, only one additional start is allowed after the pump has been running for a minimum of 10 minutes.

C. Two consecutive starts allowed, a third start is allowed after the pump has been idle for a minimum of 60 minutes.

D. After the first start, only one additional start is allowed after the pump has been idle for a minimum of 5 minutes.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because third restart is allowed if pump has RUN for 30 minutes.
B. Incorrect. Plausible because two consecutive restarts are allowed. It is not a REQUIREMENT to run the pump for any time for the second start.
C. Correct
D. Incorrect. Plausible two consecutive restarts are allowed. It is not a REQUIREMENT for the pump to be idle for any time for the second start.

Technical References: 2-SOP-24.1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-SW001 – 6

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 8
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078000K201	
		Knowledge of bus power supplies to the following: - Instrument air compressor	
	Importance	2.7	

Question # 53

Given:

A unit trip has occurred from 25% power
A fault occurred on the Station Aux Transformer
21 Diesel Generator started but the output breaker failed to automatically close.

Which of the following correctly identifies the air compressors available to supply instrument and station air?

A. 11 and 12 SAC (Centac), 21 & 22 Instrument Air Compressors

B. 21 & 22 Instrument Air Compressors, Station Air Compressor

C. 11 and 12 SAC (Centac), 22 Instrument Air Compressor

D. 22 Instrument Air Compressor, Station Air Compressor

Answer: C

Explanation/Justification:
11 SAC (Centac air compressor) is power Unit 1 buses 11SA1
12 SAC (Centac air compressor) is power Unit 1 buses 12SA2
21 Instrument Air compressor is powered from MCC29A (from Bus 5A)
22 Instrument Air compressor is powered from MCC24A (from Bus 2A)
Station Air Compressor is powered from bus 5A
21 EDG supplies bus 5A. Without bus 5A, 21 Instrument Air and the Station Air Compressors are not available.

A. Incorrect
B. Incorrect
C. Correct
D. Incorrect

Technical References: COL-29.3 and 27.1.5

Proposed References to be provided:	None		
Learning Objective:	I2LP-ILO-SA01 – 6 I2LP-ILO-SA01 – 5		
Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:	X	
	Comprehension or Analysis:		
10 CFR Part 55 Content:	55.41	(b) 4	
	55.43	(b)	
Comments:			

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078000A301	
		Ability to monitor automatic operation of the IAS, including: - Air pressure	
	Importance	3.1	

Question # 54

Given the following plant conditions:

- Plant is in cold shutdown with RCS depressurized
- RHR cooling is in service, vessel level is at 68'
- Vessel head de-tensioning in progress
- An instrument air line ruptures in the AFB: IA header pressure is 65 psig and decreasing
- Crew enters 2-AOP-AIR-1, Air System Malfunctions

Which ONE of the following statements is correct in regards to the status of the RCS?

A. No effect(s) on the RCS given the break location; a check valve in the IA supply line will effectively isolate the AFB from the rest of the station.

B. RCS level will increase without operator action due to letdown isolation and charging pump speed increase.

C. RCS level will decrease because charging line isolation valves 204A & B will fail closed and HCV-133, RHR letdown isolation valve, fails open.

D. RCS temperature is going to increase due to the isolation of CCW to the RHR heat exchangers.

Answer: B

Explanation/Justification:

A. Incorrect. Plausible because a pressure regulating valve could perform this function; howver, no such valve exists in the supply line to the AFB.
B. Correct. Candidate should know the fail position of major valves/components.

C. Incorrect. Candidate should know the fail position of major valves/components. 204A & B fail open, HCV-133 fails closed.

D. Incorrect. Candidate should know the fail position of major valves/components. 822 valves are MOVs.

Technical References:	2-AOP-AIR-1
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-SA01 – 14

Question Source:	Bank #	X	IPEC Bank - 5491
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 4
	55.43	(b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103000K406 Knowledge of Containment System design feature(s) and/or interlock(s) which provide for the following: - Containment isolation system	
	Importance	3.1	

Question # 55

Given the following conditions:

- Unit 2 is in a refueling outage
- The Containment Purge system is in service to reduce gas concentration in the Vapor Containment
- An inadvertent Safety Injection actuation occurs

Which ONE of the following describes the response of the Containment Purge System?

A. Because the SI trip is blocked, the SI actuation signal has no effect on the Containment Purge system.

B. Containment Purge supply and exhaust valves close if high containment radiation AND SI actuation signals are received.

C. Containment Purge exhaust fan only trips due to the SI actuation signal.

D. Containment Purge supply and exhaust valves close due to the SI actuation signal.

Answer: D

Explanation/Justification:

A. Incorrect. Plausible because student must recognize that the question states that an inadvertent Safety Injection actuation occurs, and not that a signal is generated. Only automatic SI is blocked during refueling. An SI signal will cause a Containment Phase A Isolation which will cause a containment ventilation isolation.

B. Incorrect. Plausible because containment purge valves will close on either signal. It does not take both to cause a ventilation isolation actuation.

C. Incorrect. Plausible because the containment purge SUPPLY fan will trip when the valves close. The exhaust fan will trip on an SI load shed. It is not the only fan that trips.

D. Correct. The SI signal will generate a containment isolation Phase A signal which will close the valves.

Technical References: System Description 10

Proposed References to be provided: None

Learning Objective: I2LP-ILO-ESS001 – 5

Question Source: Bank # X IPEC Bank

Modified Bank #

New

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge:

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 7

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	002000A201	
		Ability to (a) predict the impacts of the following malfunctions or operations on the RCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Loss of coolant inventory	
	Importance	4.3	

Question # 56

Given:

- A SBLOCA occurred leading to a reactor trip and safety injection.
- SI termination criteria were satisfied and the crew is evaluating if letdown can be re-established when it is observed that PZR level is 14% and slowly lowering.
- 21 Charging Pump is running in manual at maximum speed.

Based on these conditions what is the appropriate action to take?

A. Manually start SI Pumps as necessary to restore level and go to E-1, Loss of Primary or Secondary Coolant.

B. Manually start SI Pumps as necessary to restore level and go to ES-1.2, Post-LOCA Cooldown and Depressurization.

C. Manually actuate SI and go to E-0, Reactor Trip or Safety Injection.

D. Manually actuate SI and go to E-1, Loss of Primary or Secondary Coolant.

Answer: A

Explanation/Justification:

A. Correct per the foldout of ES-1.1

B. Incorrect but plausible. The team will be eventually going to ES-1.2 to perform an SI reduction, but E-1 is entered first.

C. Incorrect but plausible. This action is the action for similar indications in other EOPs (e.g. ES-0.1)

D. Incorrect but plausible. This action is very similar to what is specified in the procedure, however, you do not want to push SI buttons because it strips loads.

Technical References:	2-ES-1.1 Foldout page
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-EOPS11 – 2

Question Source:	Bank #		IPEC Bank
	Modified Bank #	McGuire 2003	Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 10
	55.43	(b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	015000K201 Knowledge of bus power supplies to the following: - NIS channels, components, and interconnections	
	Importance	3.3	

Question # 57

The following conditions exist:

- A plant startup is in progress.
- Reactor power is currently 7%.
- A loss of Instrument Bus 22 occurs.

Which ONE of the following describes the effect on the plant?

A. Source Range instruments energize prematurely.

B. Reactor trips due to loss of one Source Range instrument.

C. Reactor trips due to loss of one Intermediate Range instrument.

D. Intermediate Range high flux reactor trip will NOT actuate if required.

Answer: C

Explanation/Justification:

A. Incorrect. The logic to re-energize the SR NIs is 2 of 2 IR < P-6 setpoint. Plausible the candidate must remember that both IR NIs must be < P-6.

B. Incorrect. With the P-6 block still in tact, the SR tips are bypassed. Plausible because N31 is powered from IB 21 and if operating in the SR, loss of power to N-31 would cause a Rx Trip.

C. Correct. IB 21 supplies one channel of IR NIS (N-35). Loss of power to the channel will result in loss of power to protection bistables. The IR trip is a 1 of 2 coincidence; thus causing the trip.

D. Incorrect. Plausible because the candidate must understand that both control and instrument power are lost to the IR channel when IB-21 is de-energized. Furthermore the candidate must understand what effect de-energizing bistable relays has on Reactor Protection.

Technical References:	2-SOP-13.1
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-ICESC – 7 I2LP-ILO-ICESC – 9

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	011000A102	
		Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR LCS controls including: - Charging and letdown flows	
	Importance	3.3	

Question # 58

The following conditions exist:

- The RCS is being taken solid during a cooldown.
- Cooldown rate is approximately 50°F/hr
- 24 RCP is in operation
- Actual Pressurizer Level is 90% and slowly rising
- Pressurizer Pressure is 350 psig and stable

At 95% Pressurizer level the cooldown rate is reduced to 30°F/hr. How will this effect Pressurizer fill rate and what actions must be taken?

A. Pressurizer fill rate will increase.
Decrease charging pump speed

B. Pressurizer fill rate will increase
Reduce PCV-135 auto setpoint

C. Pressurizer fill rate will decrease
Increase charging pump speed

D. Pressurizer fill rate will decrease
Increase PCV-135 auto setpoint

Answer: A

Explanation/Justification:

A. Correct. Reducing cooldown rate will reduce rate of contraction of coolant and increase the fill rate. Charging flow must be reduced to remain within procedure guidelines.

B. Incorrect. Plausible because reducing cooldown rate will reduce rate of contraction of coolant and increase the fill rate. Reducing PCV-135 setpoint will increase letdown flow; however it will also reduce RCS pressure needed to maintain RCS in operation. The procedure directs only adjustment in charging flow.

C. Incorrect. Plausible because the candidate must understand the effects of changing cooldown rates on the fill rate in the pressurizer. In addition, increasing charging flow would be appropriate if the fill rate was reduced; however it is not correct since fill rate will actually be increased.

D. Incorrect. Plausible because the candidate must understand the effects of changing cooldown rates on the fill rate in the pressurizer. In addition, increasing the setpoint on PCV-135 would reduce the letdown flow which would be plausible if the fill rate was reduced; however, it is not correct since fill rate will actually increase.

Technical References: 2POP-3.3 LP I2LP-ILO-POP002

Proposed References to be provided: None

Learning Objective: I2LP-ILO-POP002 – 1

Question Source: Bank # ______ IPEC Bank

Modified Bank # ______ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ______

10 CFR Part 55 Content: 55.41 (b) 5

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	033000K402	
		Knowledge of Spent Fuel Pool Cooling System design feature(s) and/or interlock(s) which provide for the following: - Maintenance of spent fuel cleanliness	
	Importance	2.5	

Question # 59

Core offload is in progress at Unit 2. The Spent Fuel Pool cleanup system was in a normal pre-outage lineup at the start of the offload. As the offload progressed, it became necessary to place filters in the Spent Fuel Pit Temporary Cooling System (SFPTCS) in service using 2-OSP-4.3.1. What is the most likely reason that this additional filtration system had to be placed in service?

A. Increasing pool temperature leads to an increase in thermal currents. This causes silica material on the bottom of the pool to become suspended solids, so clarity is degraded.

B. Increasing pool temperature led to higher solubility of the existing suspended solids in the pool. The filters in the SFPTCS system are better than the normal filter at removing soluble suspended solids.

C. Increasing pool temperature leads to improved purification resin efficiency for boron removal. Placing the SFPTCS system filters in service allows reducing flow in the normal cleanup loop to maintain proper boron concentration.

D. Increasing pool temperature led to significantly lower purification resin efficiency in the normal cleanup loop, so clarity would be degraded without supplemental filtration.

Answer: A

Explanation/Justification:

A. Correct. This is a particular issue at IP2 because of particles on bottom of pool for boron plates in racks Temperature goes up and causes thermal currents that stirs up debris. This led to IP2 having to install a filtration system in the supplemental SFP cooling loop.
B. Incorrect but plausible. Solubility is affected by temperature so this is plausible however the debris that affects clarity is not soluble.
C. Incorrect but plausible. Temperature affect resin efficiency, however boron concentration has nothing to do with this concern.
D. Incorrect but plausible. Temperature affect resin efficiency, but soluble particles are not what degrades clarity.

Technical References: IPEC OE

Proposed References to be provided: None

Learning Objective: I2LP-ILO-SFP001 – 2

Question Source: Bank # IPEC Bank

Modified Bank # Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ~~X~~

10 CFR Part 55 Content: 55.41 (b) 7

55.43 (b) 13

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001000A306	
		Ability to monitor automatic operation of the CRDS, including: - RCS temperature and pressure	
	Importance	3.9	

Question # 60

Given the following conditions:

- Reactor power is 90%
- Control Bank D is at 200 steps
- Automatic rod control is selected

Which ONE of the following statements describes the response of the rod control system if Tavg becomes 4.5°F more than Tref? (Assume no power mismatch effects)

A. The control rods step in at 32 steps per minute.

B. The control rods step in at 48 steps per minute.

C. The control rods step in at 56 steps per minute.

D. The control rods step in at 62 steps per minute.

Answer: C

Explanation/Justification:
Rod Speed is 8 steps per minute (1.5 degree error to 3 degree error)
From 3 degrees to 5 degrees Rod Speed increases from 8 step per minute(SPM) to 72 SPM.

A. Incorrect. Plausible because this is the change in steps per minute per degree from 3 degree error signal to 5 degree error signal.
B. Incorrect. Plausible because this is the change from 8 to 72 SPM if the candidate neglects to add the starting 8 SPM.
C. Correct

D. Incorrect. Plausible because this is the change if the candidate neglects to subtract the original 8 SPM from 72 to calculate the change from 3 degree error to 5 degree error.

Technical References:	I2LP-ILO-ICROD
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-ICROD – 8

Question Source:	Bank #	X	
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035000K601	
		Knowledge of the effect of a loss or malfunction of the following will have on the S/Gs: - MSIVs	
	Importance	3.2	

Question # 61

Given the following:

- Unit 2 was operating at approximately 17% power during a power increase.
- Rod Control is in manual.
- 24 MSIV failed closed.

Which of the following identifies the plant response to this event after approximately 10 minutes? Assume no operator action.

	24 SG pressure	Turbine	Reactor	PRZR Level
A.	Lower	Tripped	Tripped	Lower
B.	Higher	Tripped	Not Tripped	Higher
C.	Higher	Not Tripped	Not Tripped	Higher
D.	Lower	Not Tripped	Not Tripped	Lower

Answer: B

Explanation/Justification:

The reactor will NOT trip with a turbine trip below P-8 (18% power). With Rod Control in Manual, rods will not step in to reduce Average Tavg. The Steam Dumps will open and remain open. Loop 24 temperature will rise to approximately Thot.

A. Incorrect. 24 SG Pressure will increase due to higher Loop Tavg in 24 loop. The turbine will trip due to the MSIV closure. The reactor will not trip. Pressurizer level will be higher due to increase in Average Tavg.
B. Correct. With loop 24 at Thot, the SG pressure will increase. The turbine will trip due to the MSIV closure. The reactor will not trip. Pressurizer level will be higher due to increase in Average Tavg.
C. Incorrect. 24 SG Pressure will increase due to higher Loop Tavg in 24 loop. The turbine will trip due to the MSIV closure. The reactor will not trip. Pressurizer level will be higher due to increase in Average Tavg.
D. Incorrect. 24 SG Pressure will increase due to higher Loop Tavg in 24 loop. The turbine will trip due to the MSIV closure. The reactor will not trip and power will remain approximately the same on the steam dumps. Pressurizer level will be higher due to increase in Average Tavg.

Technical References: 2-E-0

Proposed References to be provided: None

Learning Objective: I2LP-ILO-MTG001 - 5

Question Source: Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 5

55.43 (b) 14

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041000K302	
		Knowledge of the effect that a loss or malfunction of the SDS will have on the following: - RCS	
	Importance	3.8	

Question # 62

The plant is operating at 100 percent power when one (1) High Pressure (HP) Condenser Steam Dump valve fails full open.

Which ONE (1) of the following statements best describes the expected plant response with NO operator action?

A. An Over Temperature Delta Temperature reactor trip will occur

B. A Feed Flow / Steam Flow Mismatch reactor trip will occur

C. The plant will stabilize; 100 percent reactor power and less than 100 percent turbine power

D. The plant will stabilize; greater than 100 percent reactor power and Tave less than programmed Tave

Answer: D

Explanation/Justification:
Justification: UFSAR 14.1.11, Excessive Load Increase Incident, describes an event in which "a rapid increase in the steam flow that causes a power mismatch between reactor core power and the team generator load demand." For all cases evaluated, the UFSAR states, "the plant rapidly reaches a stabilized condition at the higher power level." In addition, Tave will decrease slightly to add (+) reactivity to compensate for the power defect. Answer D correctly states these conditions, and is the correct choice.

A. Incorrect because the OT/DT setpoint is based on not exceeding DNBR, and the UFSAR states fhat for a 10% step load increase, (one steam dump valve is app. 4% steam flow), the DNBR remains above the safety analysis limit DNBR value.

B. incorrect because the Feed/Steam flow mismatch trip is in coincidence with S/G low level.

C. incorrect because turbine load will not change, so the added steam demand will cause reactor power to increase.

D. Corrct

Technical References:	System Description 18.0 UFSAR 14.1.11
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-SDSHP – 9

Question Source:	Bank #	X	IPEC Bank 5118
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 5
	55.43	(b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	071000A426	
		Ability to manually operate and/or monitor in the control room: - Authorized waste gas release, conducted in compliance with radioactive gas discharge permit	
	Importance	3.1	

Question # 63

A Gas Decay Tank release is planned. Which of the following identifies who can authorize the release at the specified limits?

	Annual Average Limit	Quarterly Limit	Instantaneous Limit
A.	CRS	SM	Site OM
B.	CRS	SM	GMPO
C.	RO	CRS	SM
D.	RO	Site OM	GMPO

Answer: D

Explanation/Justification:
SOP-5.2.4 Calculation and Recording or Radioactive Gaseous Releases
Precaution and Limitation 2.6 identifies permission required for release below
Annual Avg - RO, CRS, SM
Quarterly Average Limit - Site Ops Manager
Instantaneous Limit - General Manager Plant Operations.

A. Incorrect. Plausible because the CRS can approve the Annual Limit; however the SM cannot approve the Quarterly Limit and the Site OM cannot approve the Instantaneous Limit

B. Incorrect. Plausible because the CRS can approve the Annual Limit; however the SM cannot approve the Quarterly Limit BUT the GMPO can approve the Instantaneous Limit

C. Incorrect. Plausible because the RO can approve the Annual Limit; however the CRS cannot approve the Quarterly Limit and the SM cannot approve the Instantaneous Limit

D. Correct

Technical References: 2-SOP-5.2.4

Proposed References to be provided: None

Learning Objective: I2LP-ILO-GWR001 – 5

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 13
	55.43	(b) 4

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	072000K101	
		Knowledge of the physical connections and/or cause-effect relationships between the ARM system and the following systems: - Plant ventilation systems	
	Importance	3.1	

Question # 64

Given the following plant conditions:

- Unit 2 is operating at 100% power.
- A containment pressure relief is in progress.
- A small leak develops inside containment on 342, Loop 21 Letdown Stop Valve, bonnet.

Which ONE of the following identifies the radiation monitor(s) that could have initiated the Containment Vent Isolation (CVI) signal, and the expected radiation monitor(s) response after the CVI?

Note the following nomenclature:

R-41, Containment Particulate
R-42, Containment Gas
R-44, Plant Vent Gas & Iodine

	Radiation Monitor	**Radiation Monitor Readings after the CVI**
A.	R-41 OR R-42 only.	R-41, R-42 and R-44 would decrease.
B.	R-44 only.	Only R-44 would decrease.
C.	R-41 OR R-42 OR R-44.	R-41, R-42 and R-44 would decrease.
D.	R-41 OR R-42 OR R-44.	Only R-44 would decrease.

Answer: D

Explanation/Justification:

Any of the 3 radiations monitors, R-41/42 and R-44 will cause a Containment Ventilation Isolation. R-41/42 sample containment atmosphere and the leak is not terminated, the response on these monitors will not decrease. R-41/42 Isolate on a SI signal and indications do decrease for that condition.

R-44 samples the Plant Discharge Duct. Since it is located downstream of Purge System, it will decrease after the CVI terminates the Containment Purge.

A. Incorrect. Plausible because R-41/42 cause CVI, but R-44 will also cause CVI. Only R-44 indication will decrease after CVI
B. Incorrect. Plausible because R-44 cause CVI, but R-41/42 will also cause CVI. Only R-44 indication will decrease after CVI
C. Incorrect. Plausible because all 3 Radiation monitors cause CVI. Only R-44 indication will decrease after CVI
D. Correct

Technical References: System Description 12.0, 2-SOP-12.3.3

Proposed References to be provided: None

Learning Objective: I2LP-ILO-RMS001 – 3

Question Source: Bank # ______ IPEC Bank

Modified Bank # Watts Bar 2009 — Note changes or attach parent

New ______

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ______

10 CFR Part 55 Content: 55.41 (b) 11

55.43 (b) 4

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	045000K523	
		Knowledge of the operational implications of the following concepts as they apply to the MT/G System: - Relationship between rod control and RCS boron concentration during T/G load increases	
	Importance	2.7	

Question # 65

Given the following:

- Reactor power was reduced to 70% to perform a repair to one Main Boiler Feedpump.
- Repairs took five days.
- A reactivity plan was developed to return the reactor to 100% in three hours.
- The plan assumed that Control Bank D would be at 180 steps at the start of the power ascension.
- Actual rod position was 200 steps on Control Bank D at the start of the power ascension.

How does this difference in rod position affect dilutions required to return to full power with rods at the normal full power position, and how will Xenon affect the power ascension?

A. Greater total dilution will be required. Dilutions amounts will have to account for increasing Xenon concentration during the power ascension.

B. Greater total dilution will be required. Dilutions amounts will have to account for the initial drop of Xenon concentration at the start of the power ascension.

C. Less total dilution will be required. Dilutions amounts will have to account for increasing Xenon concentration during the power ascension.

D. Less total dilution will be required. Dilutions amounts will have to account for the initial drop of Xenon concentration at the start of the power ascension.

Answer: B

Explanation/Justification:
Since rods are farther out than the original plan, boron concentration has to be higher than the original plan. Since boron is higher at the start,greater total dilution is required. Xenon will initially drop when power is raised. This provides another choice. The wrong answers are plausible because it is possible to get confused with how rods would affect boron concentration. Also, eventually Xenon concentration goes up to reach 100% equilibrium, so an operator could confuse that as well.

A. Incorrect
B. Correct
C. Incorrect
D. Incorrect

Technical References:

Proposed References to be provided: None

Learning Objective: I2LP-ILO-MTG001 – 5

Question Source: Bank # ______ IPEC Bank

Modified Bank # ______ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 1 , 5

55.43 (b) 6

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	Generic	
	Group #	Equip Control	
	K/A #	1940012201	
		Equipment Control - Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	
	Importance	4.5	

Question # 66

Plant conditions:

- Plant startup in progress
- Rods are 500 pcm above estimated critical position and Reactor is not critical

From the list below, identify the ONE statement that describes required action(s) for this condition.

A. Maintain rods at current position and perform a re-evaluation of inputs and mathematics used to determine ECP. Obtain Reactor Engineer's approval to resume approach to ciriticality.

B. Fully insert all control banks and perform a re-evaluation of inputs and mathematics used to determine ECP.

C. Manually trip the Reactor and perform a re-evaluation of inputs and mathematics used to determine ECP.

D. Maintain rods at current position and perform a re-evaluation of inputs and mathematics used to determine ECP. Obtain Operations Manager's approval to resume approach to criticality.

Answer: B

Explanation/Justification:
Per POP-1.2 Att 2, the correct response is to fully insert control banks and then evaluate inputs and math of ECP. There are additional actions as well, but these do not figure in the question choices.

A. Incorrect but plausible. The direction to insert control banks is conservative. It is plausible that the procedure could have us leave rods as is while evaluation takes place.
B. Correct
C. Incorrect but plausible. Tripping the Reactor is more extreme than inserting control banks, but it is plausible.
D. Incorrect but plausible. Ultimately the procedure will have us resume approach to criticality with OM permission, but not without first inserting controling banks.

Technical References: 2-POP-1.2 ATT. 2

Proposed References to be provided: None

Learning Objective:

Question Source: Bank # ______ IPEC Bank

Modified Bank # TMI 2003 Note changes or attach parent

New ______

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ______

10 CFR Part 55 Content: 55.41 (b) 6. 10

55.43 (b) 6

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	Generic	
	Group #	Conduct of Ops	
	K/A #	1940012142	
		Conduct of Operations - Knowledge of new and spent fuel movement procedures.	
	Importance	2.5	

Question # 67

Which ONE of the following is the responsibility of the ATC during core re-load?

A. Monitor source range count rate during core reload, and remain cognizant of 1/M plot results.

B. Maintain continuous communications with the Refueling Floor and Outage Control Center.

C. Maintain a 1/M plot during fuel shuffle.

D. Update the Fuel Tracking Software for each core alteration as it is performed.

Answer: A

Explanation/Justification:

The K/A is for conduct of operations and knowledge of refueling procedures. A Reactor Operator is used during fuel movement as the Refueling Monitor. This individual (who can actually be licensed on the other unit) is not part of the control room watch team.

A. Correct
B. Incorrect but plausible. An RO does this, but not the watch ROs
C. Incorrect but plausible. 1/M is maintained by the Refueling Monitor and Reactor Engineer, not the watch ROs
D. Incorrect but plausible. This is also often done by the RO who is part of refueling group

Technical References: EN-OP-115

Proposed References to be provided: None

Learning Objective: I2LP-ILO-FHD001 - 18

Question Source:	Bank #		IPEC Bank
	Modified Bank #	Beaver Valley 2005	Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 10
	55.43	(b) 7

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	Generic	
	Group #	Conduct of Ops	
	K/A #	1940012103	
		Conduct of Operations - Knowledge of shift or short-term relief turnover practices.	
	Importance	3.7	

Question # 68

Given the following plant conditions:

- The Unit is in Mode 2 following a refueling outage.
- A reactor startup is in progress IAW 2-POP-1.2, Reactor Startup.
- Due to delays in the startup, the on-coming shift has arrived in the control room for shift relief.
- Control Bank C is at 50 steps and counts are stable.

Which ONE of the following is correct concerning shift turnover IAW OAP-002, Shift Relief and Turnover?

A. Turnover can occur at any stable point (e.g., a doubling) during the start-up with the approval of the Shift Manager.

B. Turnover during the approach to criticality shall be avoided. The shift can be turned over when the startup is complete or reactor placed in a stable condition.

C. Turnover during the approach to criticality should be avoided. The shift can be turned over ONLY with the approval of the General Manager Plant Operations.

D. Turnover can occur at any stable point (e.g., a doubling) during the startup as long as NO other evolutions are in progress.

Answer: B

Explanation/Justification:

Step 4.1.3 of OAP-002 states:
Shift turnover SHALL NOT be conducted during plant transients or during major steps of an evolution (i.e., significant load changes, etc.).
Step 4.1.9 of OAP-002 states:
IF Reactor startup is in progress, THEN watch releif in CCR SHALL NOT begin until the startup is compled or the Reactor is placed in a safe stable condition

A. Incorrect
B. Correct
C. Incorrect
D. Incorrect

Technical References: OAP-002

Proposed References to be provided: None

Learning Objective: I0LP-ILO-ADM01 – 1

Question Source: Bank # ______ IPEC Bank

Modified Bank # ______ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ______

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b)

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	Generic	
	Group #	Equipment Control	
	K/A #	1940012207	
		Equipment Control - Knowledge of the process for conducting special or infrequent tests.	
	Importance	2.9	

Question # 69

Which ONE of the following surveillance tests is required to be designated as an Infrequently Performed Test or Evolution?

A. 2-PT-2M5, Safety Injection System Train A Actuation Logic and Master Relay Test

B. 2-PT-SA067, Cable Spread Halon System

C. 2-PT-2Y008A, 21 EDG Mechanical Overspeed Trip

D. 2-PT-Q48, AMSAC Logic

Answer: C

Explanation/Justification:

This question is fair from memory because an operator should know what the entry conditions of the procedure. Actual requirements of the procedure would not be fair from memory.

A. Incorrect. Plausible because this test has the potential to cause a reactor trip; however it does not meet the guidance in EN-OP-116 primarily the test is performed more frequently than quarterly and is covered by an existing approved procedure.

B. Incorrect. Plausible because this test has a potential safety and equipment inoperability risk; however it does not meet the guidance in EN-OP-116 primarily the test is covered by an existing approved procedure.

C. Correct. EN-OP-116 states any test that actually overspeeds a turbine or Emergency Diesel Generator.

D. Incorrect. Plausible because this test has the potential to cause a reactor trip; however it does not meet the guidance in EN-OP-116 primarily the test is performed more frequently than quarterly and is covered by an existing approved procedure.

Technical References:	EN-OP-116
Proposed References to be provided:	None
Learning Objective:	I0LP-ILO-ADM01 – 1

Question Source:	Bank #		IPEC Bank
	Modified Bank #	North Anna 2008	Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 5
	55.43	(b) 10

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	Generic	
	Group #	Equipment Control	
	K/A #	1940012222	
		Equipment Control - Knowledge of limiting conditions for operations and safety limits.	
	Importance	4.0	

Question # 70

The plant is at 100% power when the following ECCS accumulators become inoperable:

- 0230 on July 2, number 2 ECCS accumulator is declared inoperable due to boron concentration of 1950 ppm.
- 1500 on July 3, number 4 ECCS accumulator is declared inoperable due to a volume of 880 ft3.

Which one of the following describes the time number 2 accumulator is required to be restored to OPERABLE status without requiring entry into a plant shutdown condition? The number 2 ECCS accumulator must be restored to OPERABLE status by:

A. July 3 at 1500 hours.

B. July 3 at 1600 hours.

C. July 4 at 1500 hours.

D. July 5 at 0230 hours.

Answer: A

Explanation/Justification:

A. Correct. TS 3.0.3 is entered immediately which requires a plant shutdown.
B. Incorrect. Plausible because the candidate may believe that TS 3.0.3 allows one hour to restore the accumulator before shutdown is required.

C. Incorrect. Plausible because candidate may believe that the shutdown action is initiated when the shorter 24 completion time for #4 accumulator expires.

D. Incorrect. Plausible because the candidate may believe that separate entry conditions apply for each accumulator. This date represents the expiration of the initial 72 AOT for #2 accumulator.

Technical References:

Proposed References to be provided: Tech Spec 3.5.1

Learning Objective: I2LP-ILO-SIS001-11

Question Source: Bank # X IPEC Bank 2443

Modified Bank # ____ Note changes or attach parent

New ____

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ____

10 CFR Part 55 Content: 55.41 (b) 3

55.43 (b) 2

Comments:

Exam Outline Cross Reference:	Level	<u>RO</u>	<u>SRO</u>
	Tier #	Generic	
	Group #	Rad Controls	
	K/A #	1940012305	
		Radiological Controls - Ability to use radiation monitoring systems, such as fixesd radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	
	Importance	2.9	

Question # 71

Given the following conditions:

- The plant is at 100% power
- 23 Large Gas Decay Tank is aligned for in-service and re-use
- 24 Large Gas Decay Tank is in standby
- 22 Large Gas Decay Tank is isolated with a pressure of 90 psig and a content of 5000 Curies
- All remaining Gas Decay Tanks are inerted with nitrogen
- 22 Large Gas Decay Tank relief valve (1622) fails open
- No radiation monitors were in alarm prior to the 1622 failure

Which ONE of the following describes the plant response to this event?

A. High radiation level alarm on R-50, Waste Gas Decay Tank Monitor AND R-44, Plant Vent Air Monitor.

B. High radiation level alarm on R-44, Plant Vent Air Monitor. R-50, Waste Gas Decay Tank Monitor does NOT alarm.

C. High radiation level alarm on R-50, Waste Gas Decay Tank Monitor. R-44, Plant Vent Air Monitor does NOT alarm.

D. NO high radiation level alarm on R-50, Waste Gas Decay Tank Monitor OR R-44, Plant Vent Air Monitor.

Answer: B

Explanation/Justification:

For this situation, the tank will relieve directly to the plant vent and be monitored by R-44. R-50 should be unaffected by this leak.

A. Incorrect but plausible. Plausible because a candidate may wrongly assume that R-50 would go up
B. Correct
C. Incorrect but plausible. Plausible because a candidate may believe R-50 would go up and may have a misconception of where this relieves to (e.g. WHUT) or that R-44 will not alarm.
D. Incorrect but plausible. Plausible because a candidate may believe neither monitor will alarm or may think it relieves to a closed tank.

Technical References:	System Description 12.0 2-SOP-12.3.3
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-GWR01 – 13

Question Source:	Bank #	X	IPEC Bank 8370
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 12, 13
	55.43	(b) 4

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	Generic	
	Group #	Rad Controls	
	K/A #	1940012312	
		Radiological Controls - Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	
	Importance	3.2	

Question # 72

Unit 2 is at 1% Reactor power coming out of a refueling outage.

Personnel are in containment making adjustments to 23 RCP vibration probes.

The CRS and SM decide they want to raise power to 2% in preparation for power ascension later that day.

Based on OAP-007, Containment Entry and Egress, what is required regarding this power ascension?

A. Personnel working on the RCP vibration probes will have to move to the outer crane wall. When the power increase is complete the workers can return to the RCP.

B. Power can be raised. Since the plant will remain in a mode below Mode 1 dose rate changes will be minimal, so the power ascension does not require additional action per OAP-007.

C. Power can be raised. However, since there are personnel in the inner crane wall, OAP-007 requires the SM to specifically approve the power ascension.

D. The RP Supervisor and entry party must be notified prior to any planned change in power level. The RP Supervisor will then decide if workers need to exit or move to ALARA area prior to raising power if necessary.

Answer: D

Explanation/Justification:
This situation actually occurred at IP3, which led to the procedural requirement.

A. Incorrect but plausible. It is not unreasonable that OAP-007 would have required removing personnel prior to power ascension, not just moving to outer crane wall.
B. Incorrect but plausible. It would be reasonable to assume that this power change would have minimal effect on dose rates, but this is not true.
C. Incorrect but plausible. The SM is often allowed to authorize items that require slightly greater levels of control and decision making. Based on B above discussion, it is reasonable that a candidate may assume this change will have minimal effect.
D. Correct based OAP-007 step 2.22

Technical References: OAP-007

Proposed References to be provided: None

Learning Objective: I0LP-ILO-ADM01 – 4

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 10
	55.43	(b) 5, 6

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	Generic	
	Group #	Emergency Procedures/Plan	
	K/A #	1940012421	
		Emergency Procedures/Plan - Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	
	Importance	4	

Question # 73

The STA is monitoring the Critical Safety Function Status Trees. Which of the following sets of parameters would cause the implementation of FR-P.1, Response to imminent Pressurized Thermal Shock condition?

A.
- Cold Leg temperatures decreased from 540°F to 310°F in the last hour
- RCS pressure is 1500 psig

B.
- Cold Leg temperatures decreased from 450°F to 360°F in the last hour
- RCS pressure is 800 psig

C.
- Cold Leg temperatures decreased from 540°F to 280°F in the last hour
- RCS pressure is 600 psig

D.
- Cold Leg temperatures decreased from 370°F to 290°F in the last hour
- RCS pressure is 900 psig

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because the temperature decrease was greater than 100 degrees in the last 60 minutes, RCS pressure is relatively high and temperature is relatively cool. This set of conditions will direct you to FR-P.2 not FR-P.1.

B. Incorrect. Plausible because FR-P.1 can be entered due to either a Pressurized Thermal Shock condition (i.e., > 100 degree cooldown in the last hour combined with low temperature and high pressure) or a Cold Overpressure Condition (low temperature combined with high pressure). The cooldown is < 100 degrees in the last hour, but the pressure is relatively high. The temperature does no meet the Cold Overpressurization limit for entry into FR-P.1

C. Correct. The cooldown exceeded the 100 degree in the last hour requirement and the temperatures are low. While the pressure is relatively low, it does meet the ORANGE path terminus conditions. While these conditions are not as sever as a RED path terminus, the actions are directed by FR-P.1.

D. Plausible because this represents a potential Cold Overpressure Condition which can result in a transition to FR-P.1

Technical References: PTS Status Tree

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPFP1-2

Question Source: Bank # X IPEC Bank 18620

Modified Bank # ___ Note changes or attach parent

New ___

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis: ___

10 CFR Part 55 Content: 55.41 (b) 5

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	Generic	
	Group #	Emerg Proc/Plan	
	K/A #	1940012419	
		Emergency Procedures/Plan - Knowledge of EOP layout, symbols, and icons.	
	Importance	3.4	

Question # 74

Given the following conditions:

- The crew is responding to a large break LOCA
- A CORE COOLING status tree ORANGE path causes a transition to FR-C.2, Response to Degraded Core Cooling
- During performance of FR-C.2, the CORE COOLING status tree changes from ORANGE to YELLOW
- An ORANGE path exists on the CONTAINMENT status tree

Which ONE of the following describes the required action(s)?

A. Complete FR-C.2 and then go to FR-Z.1, because a functional restoration procedure must be completed unless preempted by a higher priority condition.

B. Go to FR-Z.1, because an ORANGE path has higher priority than a YELLOW path. Completion of FR-C.2 is not needed.

C. Go to FR-Z.1, then complete FR-C.2 because the CORE COOLING status tree had been in an ORANGE path.

D. Perform FR-C.2 and FR-Z.1 concurrently, because FR procedures of the same priority can be executed together.

Answer: A

Explanation/Justification:

A. Correct Answer: Step 4..3.13 of OAP 12 requires the completion of a FRP entered due to a RED or ORANGE condition unless that FRP is preempted by a higher priority condition.
B. Orange is higher priority than Yellow, but OAP 12 step 4.3.13 requires the completion of the current procedure.

C. FR-C.2 has higher priority than FR-Z.1 and needs to be completed first in accordance with OAP 12 step 4.3.13.

D. FR-C.2 is the higher priority and needs to be completed first in accordance with OAP 12 step 4.3.13.

Technical References:	OAP-12		
Proposed References to be provided:	None		
Learning Objective:	I2LP-ILO-EOPROU - 12		
Question Source:	Bank #	X	IPEC Bank
	Modified Bank #		Note changes or attach parent
	New		
Question History:	Last 2 NRC Exams at IPEC:	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:	X	
10 CFR Part 55 Content:	55.41	(b) 10	
	55.43	(b) 5	
Comments:			

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #	Generic	
	Group #	Emerg Proc/Plan	
	K/A #	1940012429	
		Emergency Procedures/Plan - Knowledge of the emergency plan.	
	Importance	3.1	

Question # 75

Given the following:

- A Site Area Emergency has been declared.
- The Emergency Response Organization is staffed.
- A repair team consisting of 1 NPO, 1 mechanic, and 1 HP technician must be sent to the PAB to isolate a leak.

Which ONE of the following Emergency Response Facilities is responsible for assembly and preparation of the team?

A. Control Room

B. Technical Support Center (TSC)

C. Operational Support Center (OSC)

D. Emergency Operations Facility (EOF)

Answer: C

Explanation/Justification:

A. Incorrect. Make initial declaration and classification of event. Manipulation of the reactor or plant to mitigate the consequences of an accident remain the primary function of the CR. Plausible because before Emergency Response facilities are manned, the CR would direct this action.

B. Incorrect. The TSC is the central facility for the accumulation and re-transmittal of plant parameters. The TSC provides Technical Support. Plausible because candidate may confuse functions performed by different facilities.

C. Correct. The OSC is where survey, operations and repair teams are dispatched into areas of the plant and is the staging area for individual who may be assigned.

D. Incorrect. The EOF provides overall management of the Indian Point response.

Technical References: IP-EP-230

Proposed References to be provided: None

Learning Objective: I0LP-ILO-ERT001 – 1

Question Source:	Bank #		IPEC Bank
	Modified Bank #	Ginna 2007	Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 10
	55.43	(b) 5

Comments:

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date: July 22, 2010	Facility/Unit: IPEC Unit 2
Region: I ☒ II ☐ III ☐ IV ☐	Reactor Type: W ☒ CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results		
RO/SRO-Only/Total Examination Values	____ / ____ / ____	Points
Applicant's Scores	____ / ____ / ____	Points
Applicant's Grade	____ / ____ / ____	Percent

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- ERASE COMPLETELY TO CHANGE
- EXAMPLE: A B C D E

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- Mark total possible subjective points
- Only one mark per line on key
- 163 points maximum

EXAMPLE OF STUDENT SCORE:

NAME	SRO ANSWER KEY		
SUBJECT		TEST NO.	
DATE		PERIOD	

TEST RECORD	
PART 1	
PART 2	
TOTAL	

PART 1

(T) (F) KEY

% (F) 3 KEY

1	C
2	A
3	B
4	A
5	B
6	C
7	C
8	A
9	A
10	D
11	D
12	B
13	A
14	A
15	D
16	C
17	A
18	B
19	D
20	B
21	D
22	A
23	D
24	D
25	B
26	D
27	A
28	B, C
29	C
30	C
31	B
32	C
33	B
34	C
35	B
36	D
37	D
38	C
39	D
40	B
41	D
42	A
43	D
44	A
45	D
46	C
47	B
48	B
49	C
50	C

DC

50

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EXAMPLE OF STUDENT SCORE:

PART 2

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SUBJECT		TEST NO.	
DATE		PERIOD	

TEST RECORD	
PART 1	
PART 2	
TOTAL	

	(T)	(F)			KEY
	%	**2**	3	**4**	**5**
51	A	**B**	C	D	E
52	A	B	**C**	D	E
53	A	B	**C**	D	E
54	A	**B**	C	D	E
55	A	B	C	**D**	E
56	**A**	B	C	D	E
57	A	B	**C**	D	E
58	**A**	B	C	D	E
59	**A**	B	C	D	E
60	A	B	**C**	D	E
61	A	**B**	C	D	E
62	A	B	C	**D**	E
63	A	B	C	**D**	E
64	A	B	C	**D**	E
65	A	**B**	C	D	E
66	A	**B**	C	D	E
67	**A**	B	C	D	E
68	A	**B**	C	D	E
69	A	B	**C**	D	E
70	**A**	B	C	D	E
71	A	**B**	C	D	E
72	A	B	C	**D**	E
73	A	B	**C**	D	E
74	**A**	B	C	D	E
75	A	B	**C**	D	E
76	**A**	B	C	D	E
77	A	B	**C**	D	E
78	A	**B**	C	D	E
79	**A**	B	C	D	E
80	**A**	B	C	D	E
81	A	**B**	C	D	E
82	A	B	C	**D**	E
83	**A**	B	C	D	E
84	A	B	C	**D**	E
85	A	**B**	C	D	E
86	A	**B**	**C**	D	E
87	A	**B**	C	D	E
88	**A**	B	C	D	E
89	**A**	B	C	D	E
90	A	B	**C**	D	E
91	A	**B**	C	D	E
92	A	**B**	C	D	E
93	A	**B**	C	D	E
94	A	B	**C**	D	E
95	A	B	C	**D**	E
96	A	**B**	C	D	E
97	A	**B**	C	D	E
98	A	B	**C**	D	E
99	A	**B**	C	D	E
100	**A**	B	C	D	E

B&C Correct for Q 86
Per Post EXAM Comment
Resolution J. L. Caruso 9/1/10

FEED THIS DIRECTION

50

8 10054 00013 6

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000008A213	
		Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: - High-pressure safety injection pump flow indicator, ammeter, and controller	
	Importance		3.9

Question # 76

Given the following:

- A Pressurizer Safety Valve failed open.
- 23 SIP is out of service for maintenance.
- All other equipment functions as designed.
- RCS Temperature stabilized at approximately 530°F.
- RCS Pressure stabilized at approximately 950 psig.
- Approximately 15 minutes after the safety injection actuation 21 SIP tripped on overcurrent.
- The team has just transitioned to E-1 Loss of Reactor or Secondary Coolant.
- E-0, Reactor Trip or Safety Injection, Attachment 1 is in progress.

Which of the following correctly states the expected SI flowrate indications and procedural actions for this condition?

A. Approximately 0 gpm to each RCS loop. Establish SI flow using E-0 Attachment 1.

B. Approximately 0 gpm to each RCS loop. Do not use E-0 Attachment 1 to establish flow since E-1 has been entered and will address this condition.

C. Approximately 100 gpm to each RCS loop. The EOPs will not require any adjustments to SI flow.

D. Approximately 200 gpm each to loops 22 & 24. E-1 will allow for balancing SI flow if desired.

Answer: A

Explanation/Justification:
21 SIP supplies loops 21 & 23. 23 SIP supplies loops 22 & 24. 22 SIP supplies all loops unless 21 or 23 pump breakers are open with an SI signal present. If 21 SIP breaker is open with an SI signal present, 851B (discharge valve for 22 SIP to loops 22 & 24) will automatically close. If 23 SIP breaker is open with an SI signal present, 851A (discharge valve for 22 SIP to loops 21 & 23) will automatically close. This condition is only addressed in E-0 Attachment 1

A. Correct
B. Incorrect. Plausible because the flowrate is correct, but the condition is not addressed by E-1.
C. Incorrect. Plausible because the flowrate is correct for22 SIP; however, with both 21 & 23 SIPs tripped, 851 A/B will automatically close. Candidate may believe that 22 SIP is supplying all loops.
D. Incorrect. Plausible because the flowrate is correct for 22 SIP; however, with both 21 & 23 SIPs tripped, 851 A/B will automatically close. Candidate may believe 851B will remain open. In addition, E-1 will not address this condition.

Technical References: E-0 Attachment 1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPE00 - 3

Question Source: Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b)

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0000112408	
		Emergency Procedures/Plan - Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	
	Importance		4.5

Question # 77

Given the following conditions:

- A loss of Instrument Air has occurred.
- The CRS has directed a reactor trip in accordance with the requirements of AOP-AIR-1, Air System Malfunctions.
- When the Reactor was tripped, a Large Break LOCA occurred.
- E-0, Reactor Trip or Safety Injection has just been entered.

Which ONE of the following describes the allowable usage of AOP-AIR-1 while responding to this event?

A. When E-0 immediate actions are complete, resume AOP-AIR-1 until all actions are completed. Verification of automatic actions in E-0 cannot be performed with a loss of Instrument Air.

B. Discontinue use of AOP-AIR-1 until transition to any recovery procedure. Parallel use is only allowed when E-0 is complete.

C. When E-0 immediate actions are complete, parallel use of AOP-AIR-1 is allowed when performance will not detract from performance of EOPs.

D. Discontinue use of-AOP-AIR-1. The EOP network will direct actions to restore Instrument Air to vital components.

Answer: C

Explanation/Justification:

From OAP-015 AOP Users Guide

4.1.18 IF directed to INITIATE actions in a referenced procedure OR attachment THEN the actions should be taken while continuing on in the AOP.

4.1.18.1 IF an AOP directs the initiation of E-0, THEN the AOP actions will normally be taken after transition to ES-0.1 (Reactor Trip Response) or after step 4 of E-0, (Reactor Trip Or Safety Injection) WHEN performance will NOT detract from performance of the EOP.

4.1.18.2 The CRS may delegate the completion of AOP actions while continuing in the EOPs.

A. Incorrect. Plausible because the actions of AOP-AIR-1 may restore air pressure and make verification of auto actions proceed more smoothly.
B. Incorrect. As shown above, an AOP can be performed in parallel with an EOP. Plausible because the OAP does state that the AOP will NORMALLY be resumed when transition to ES-0.1.
C. Correct
D. Incorrect. Plausible because some AOPs direct an unconditional exit to E-0. For those AOPs, no further actions are taken in the AOP.

Technical References: OAP-015

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPROU - 19

Question Source: Bank # ____ IPEC Bank

Modified Bank # X Note changes or attach parent 16887

New ____

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000011A213	
		Ability to determine and interpret the following as they apply to a Large Break LOCA: - Difference between overcooling and LOCA indications	
	Importance		3.7

Question # 78

Given the following conditions:

- The reactor has tripped. Safety Injection and Containment Spray have actuated.
- The team is performing the actions of E-1, Loss of Reactor or Secondary Coolant
- A Red Path exists on the Integrity Status Tree
- The CRS directs transition to FR-P.1, Response to Imminent Pressurized Thermal Shock
- The procedure immediately sends the team back to E-1

Which of the following identifies the plant parameters checked to immediately exit FR-P.1 and why FR-P.1 is not implemented?

A. RCS Pressure and SG Pressure. SG pressure greater than RCS pressure indicates a Large Break LOCA vice a Steam Break; the excessive cooldown will not continue.

B. RCS Pressure and RHR Flow. RHR Flow greater than the minimum value inidicates a Large Break LOCA and thermal shock is not a serious concern for this event.

C. Containment Radiation and RHR Flow. Elevated Radiation and RHR flow above the minimum value indicate a Large Break LOCA; repressurization of the RCS is virtually impossible during a Large Break LOCA.

D. RCS Pressure and Containment Pressure. RCS Pressure and Containment Pressure approximately equal indicate a Large Break LOCA. The actions in FR-P.1 will delay the actions in ES-1.3, Transfer to Cold Leg Recirculation, causing a potential loss of core cooling.

Answer: B

Explanation/Justification:
FR-P.1 step 1 checks RCS pressure and RHR Flow to identify a Large Break LOCA.

A. Incorrect. Plausible because RCS pressure will be less than SG pressure on a large break LOCA; however, the cooldown will continue until transfer to recirc.
B. Correct. From the background document: For transients where RCS pressure is less than the RHR pump shutoff head and flow from the RHR pumps has been verified, the operator should return to the procedure and step in effect since these symptoms are indicative of a large-break LOCA. In this instance, the actions of 2-FR-P.1 should not be performed since pressurized thermal shock is not a serious concern for a large-break LOCA.
C. Incorrect. Plausible because containment radiation is the key parameter used to distinguish between a LOCA and Steam Break accident inside containment; however these are not the parameters used to identify a LBLOCA. Also, "repressurization during a LBLOCA is virtually impossible" is true.
D. Incorrect. Plausible because RCS Pressure and Containment Pressure will be approximately equal during a LBLOCA. These are not the parameters used. Also actions in ES-1.3 are time critical to establish flow before the RWST empties.

Technical References:	2-FR-P.1 Background
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-EOPFP1 – 1

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 10

	55.43	(b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000040A202	
		Ability to determine and interpret the following as they apply to the Steam Line Rupture: - Conditions requiring a reactor trip	
	Importance		4.7

Question # 79

Given the following plant conditions on Unit 2:

- Containment Pressure: 1 psig and rising.
- RCS pressure: 2225 PSIG and lowering.
- Reactor power: 63% and rising.
- Average Tavg: 557°F and lowering.
- Turbine power: 561 MWe and lowering.

Based on the above plant indications, what event is occurring and what are the required actions/procedures to address the event?

A. A Steamline Break. Trip the reactor trip, Close MSIVs and go to E-0, Reactor Trip or Safety Injection.

B. A Steamline Leak. Perform a rapid Load reduction per AOP-RLR-1, Rapid Load Reduction.

C. A Small Break RCS LOCA. Trip the reactor and go to E-0, Reactor Trip or Safety Injection.

D. An RCS Leak. Perform AOP-LEAK-1, Sudden Increase in Reactor Coolant Leakage.

Answer: A

Explanation/Justification:
Immediate Actions of AOP-UC-1 require a Reactor Trip, Close MSIVs and go to E-0 for unisolable Steam Leak.

A. CORRECT. Reactor power is rising, indicating positive reactivity event. Electric load is lowering, indicating loss of steam to the turbine. Turbine power should be closer to 670 Mw with Tavg closer to 558°F based on this reactor power. Based on this degree of mismatch and unisolable (inside VC) reactor trip is required.

B. INCORRECT. A Steam Leak is occuring, but based on the large mismatch a reactor trip is required.

C. INCORRECT. Plausible since RCS pressure is lowering, but Reactor power is rising - indicating positive reactivity event.

D. INCORRECT. Plausible since RCS pressure is lowering, but Reactor power is rising - indicating positive reactivity event.

Technical References: 2-AOP-UC-1

Proposed References to be provided: None

Learning Objective: I2LP-ILO-AOPUC1 – 1

Question Source: Bank # ____ IPEC Bank

Modified Bank # DC Cook 2007 Note changes or attach parent

New ____

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0000582446	
		Emergency Procedures/Plan - Ability to verify that the alarms are consistent with the plant conditions.	
	Importance		4.2

Question # 80

Given the following:

- A LBLOCA occurs while Unit 2 is operating at 100% power.
- Automatic Reactor Trip and Safety Injection occur.
- During the bus transfer, the Station Auxiliary Transformer trips and the transient results in the breaker connecting 24 Battery to 24 DC PP tripping open.
- There is no fault on 24 DC Bus. All other equipment operates as designed.

Assuming no operator action to re-close the breaker, how does this affect operator response to the event?

A. Panel SBF-2 alarms will lose power. RWST level will have to be closely monitored to ensure transition to ES-1.3 Cold Leg Recirculation is properly made.

B. 23 EDG will start and power Bus 6A, however safeguards loads will have to be manually started in accordance with ES-0.1 Reactor Trip Response.

C. Safeguards loads will automatically sequence on, but only after 23 EDG output breaker is manually closed in accordance with the Alarm Response Procedure.

D. Both trains of Core Exit Thermocouples will be lost. Hot leg temperatures and wide range pressure will have to be used to monitor for RCP trip criteria.

Answer: A

Explanation/Justification:

This question comes down to understanding that alarms will not be available for this fault. 6A switchgear and 23 EDG control power will automatically swap to backup DC, but the alarm power will not. It also may not be clear to a candidate that the battery charger can not supply the bus.

A. Correct. These alarms will not function. This is why how the KA is being tested because the candidate has to understand that this lack of alarm capability is consistent with plant conditions. Also there are consequences of not understanding this because backup methods of monitoring for ES-1.3 transition have to used.

B. Incorrect but plausible. Plausible because a candidate may not realize just what will automatically back up. The switchgear power will automatically swap, so this equipment will operate normally.

C. Incorrect but plausible. Plausible because a candidate may not realize just what will automatically back up. The switchgear power will automatically swap, so this equipment will operate normally.

D. Incorrect but plausible. Plausible because a candidate may not realize just what is powered from DC bus 24. Only one train of CETs will be affected.

Technical References: 2-AOP-DC-1 Attachment 12

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EDS03 – 11

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0000092125	
		Conduct of Operations - Ability to interpret reference materials such as graphs, curves, tables etc.	
	Importance		4.2

Question # 81

Given the following conditions:

- A reactor trip with SI occurred at 1240 hours
- At 1550 hours, the control room operators transitioned from E-1, Loss of Reactor or Secondary Coolant, to ECA-1.1, Loss of Emergency Coolant Recirculation
- At 1600 hours, the operators are determining if SI flow can be terminated

The following conditions are observed:

- RWST level is 15 feet
- RCS Wide Range Pressure is 1600 psig
- RCS subcooling based on core exit TCs is 70°F
- CNMT Pressure is 8 psig
- All RCPs have been secured
- RVLIS level is 69% on Natural Circulation Range

Using attached procedure, which ONE of the following actions is required?

A. Terminate Safety Injection.

B. Establish a minimum of 235 gpm Safety Injection flow.

C. Establish a minimum of 275 gpm Safety Injection flow.

D. Establish a minimum of 460 gpm Safety Injection flow.

Answer: B

Explanation/Justification:

The subcooling requirement for terminating SI is not met, so the ECA-1.1 table is used to determine flow. 200 minutes have elapsed, so 235 gpm required.

A. Incorrect but plausible. 70 degrees meets the non-adverse subcooling requirement to terminate SI, so this is plausible.
B. Correct per ECA-1.1 step 14 and figure ECA11-1
C. Incorrect but plausible. Plausible since table can be read incorrectly.
D. Incorrect but plausible. Plausible since table can be read incorrectly.

Technical References: 2-ECA-1.1

Proposed References to be provided: 2-ECA-1.1 PG 11 & 31

Learning Objective: I2LP-ILO-EOPC11 – 1
I2LP-ILO-EOPC11 – 2

Question Source:
Bank # X IPEC Bank - 8307
Modified Bank # ____ Note changes or attach parent
New ____

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level:
Memory or Fundamental Knowledge: ____
Comprehension or Analysis: X

10 CFR Part 55 Content:
55.41 (b)
55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	0000032222	
		Equipment Control - Knowledge of limiting conditions for operations and safety limits.	
	Importance		4.7

Question # 82

Reactor power is 80% during a power ascension following a forced outage. During rod motion the stationary gripper fuse blows for Rod C-3. Repairs are expected to take several days.

Which of the following statements is correct regarding continued operation with this condition?

A. Power can be held at the current level provided a flux map is performed within 12 hours to ensure core hot channel factor limits are not exceeded.

B. Power can be held at the current level provided a flux map is performed within 12 hours to ensure core hot channel factor limits are not exceeded and safety analyses are re-evaluated within 5 days to ensure they are valid for current conditions.

C. A power reduction will be required to ensure that axial flux difference limit assumptions are valid.

D. A power reduction will be required to ensure core hot channel factor limits are not exceeded.

Answer: D

Explanation/Justification:

A. incorrect but plausible. 85% is significant power level for determining how far can mis-aligned, so an operator could assume 80% is low enough to allowed continued operation. There are 12 hour T.S. requirements and a flux map is required which add to the plausibility of continued operation.

B. incorrect but plausible. See above, plus there are requirements for performing this analyses.

C. incorrect but plausible. A power reduction is required, but the the reason is not related to AFD. For this dropped rod (near a single NI) AFD is only affected in one channel.

D. Correct per T.S. and basis for LCO 3.1.4

Technical References: Tech Specs 3.1.4

Proposed References to be provided: None

Learning Objective: I2LP-ILO-ICROD – 14

Question Source: Bank # X IPEC Bank 24193

Modified Bank # Note changes or attach parent

New

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge:

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b)

55.43 (b) 2, 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	000024A202	
		Ability to determine and interpret the following as they apply to the Emergency Boration: - When use of manual boration valve is needed	
	Importance		4.4

Question # 83

Unit 2 was operating at 100% power with no equipment out of service when the following occurred:

- A loss of instrument air pressure occurred in the Primary Auxiliary Building.
- Subsequently 112C (VCT Outlet Valve) closed and cannot be re-opened.
- No other equipment failures occurred.

How do these failures affect operability of the Boration System specified in the Technical Requirements Manual for use in Emergency Boration of the Reactor?

A. The TRO is satisfied because Boration is still available from the RWST and the boric acid storage system. Emergency Boration could be performed using MOV-333 (Emergency Boration Valve).

B. The TRO is satisfied because Boration is still available from the RWST and the boric acid storage system. Emergency Boration could be performed using LCV-112B (RWST Emergency M/U Valve) which has failed open due to the loss of instrument air.

C. The TRO is not satisfied because Boration is only available from the RWST. Emergency Boration could be performed using LCV-112B (RWST Emergency M/U Valve) which has failed open due to the loss of instrument air.

D. The TRO is not satisfied because Boration is only available using MOV-333. Emergency Boration can NOT be performed using LCV-112B (RWST Emergency M/U Valve) since it has failed closed.

Answer: A

Explanation/Justification:
There are two elements to this question. One element is understanding what is needed to satisfy the TRO. TRM 3.1.B.1 requires 1 path from RWST, 1 path from BAST and 2 charging pumps. The VCT is not needed. Manual field action is allowed to meet this TRO. We can RWST and BAST water to the suction of all the charging pumps, so the TRO is satisfied. The other element of this question is understanding LCV-112B operation with a loss of air.

A. Correct
B. Incorrect because LCV-112B does not fail open. This answer is plausible because the "safe" position of LCV-112B is open, however it has backup N2 to open it if air is lost.
C. Incorrect because the TRO is satisfied and because LCV-112B does not fail open (see B)
D. Incorrect because the TRO is satisfied and because LCV-112B could be operated locally to borate.

Technical References: TRM 3.1.B.1, System Description 3.0

Proposed References to be provided: None

Learning Objective: I2LP-ILO-CVC02 – 5
I2LP-ILO-CVC02 – 14

Question Source: Bank # ______ IPEC Bank

Modified Bank # ______ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______
Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b) 2, 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	000037A202	
		Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: - Agreement/disagreement among redundant radiation monitors	
	Importance		3.9

Question # 84

Given the following:

- A startup is in progress at 20% power.
- Radiation Monitor 45 (Air Ejector Exhaust) is OOS.
- Radiation Monitor 55B (22 SG Blowdown) is in Alarm.
- Radiation Monitor 61B (22SG N-16) remains unchanged
- No other radiation monitor Alarm or Warn condition exists.

Which of the following is correct regarding these conditions?

A. Radiation Monitor 55B (22 SG Blowdown) has failed and should be declared inoperable. Radiation Monitor 61B (22 SG N-16) is the most sensitive to SG tube leakage and should indicate actual leakage before R-55B (22 SG Blowdown).

B. Radiation Monitor 55B (22 SG Blowdown) has failed and should be declared inoperable. Radiation Monitor 29 (22 Main Steam Line) should also be at the Warn or Alarm setpoint if an actual tube leak existed that caused R-55B to alarm.

C. Radiation Monitor 55B (22 SG Blowdown) may indicate a tube leak. 2-AOP-SG-1 SG Tube Leakage will use Radiation Monitor 61B (22 SG N-16) to confirm or eliminate the existence of tube leakage.

D. Radiation Monitor 55B (22 SG Blowdown) may indicate a tube leak. The lack of redundant radiation monitors does not eliminate a tube leak.

Answer: D

Explanation/Justification:
In general the lack of redundant indication at this power level does not eliminate the existence of a SG tube leak. The Warn setpoint for Radiation Monitor 45 is set for an equivalent 30 ppd (gallons per day) leakrate. The alarm setpoint for Radiation Monitors 61A-D is set for 5 gpd. However, below 30% power R-61A-D may not indicate accurately.

R-55B senses radiation in 22 SG Blowdown line. It is not nearly as sensitive as the N-16 monitors or R-45, so it very plausible that an operator may assume these indications are indicative of a failure. However, with R-45 OOS and the N-16 monitors not being sensitive below 30% power, R-55B could be the first indication of SGTL on 22 SG. AOP-SG-1 would use a chemistry sample to back up the reading.

A. Incorrect: Radiation Monitors 61A-D may not be accurate below 30% power per AOP-SG-1 Background Document.
B. Incorrect: The setpoint for Radiation Monitor 29 may not be accurate and thus cannot be used to eliminate a SG tube leak.
C. Incorrect: Radiation Monitors 61A-D may not be accurate below 30% power per AOP-SG-1 Background Document.
D. Correct

Technical References: System Description 12.0

Proposed References to be provided: None

Learning Objective: I2LP-ILO-RMS001 – 5

Question Source: Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 11

55.43 (b) 4

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	00WE102406	
		Emergency Procedures/Plan - Knowledge of EOP mitigation strategies.	
	Importance		4.7

Question # 85

The Team is performing ES-0.4, (Natural Circulation Cooldown With Steam Void in Vessel (Without RVLIS)). Step 6 "Equalize Charging and Letdown" has been performed.

What is the purpose of equalizing Charging and Letdown?

A. So a void formation in the vessel will be minimized

B. So changes in pressurizer level will be an indication of void formation

C. So the pressurizer will not go water solid

D. So letdown isolation / heater trip will not occur during depressurization

Answer: B

Explanation/Justification:

A. Incorrect but plausible. Keeping voids to a minimum is an overall goal of the procedure but these steps are not specifically to do this. Maximizing charging would minimize voids.

B. Correct per the background document for ES-0.4 in the purpose section for this step

C. Incorrect but plausible. Having excessive charging (which may be the case if operators were trying to minimize voids) could increase the likelihood of going water solid.

D. Incorrect but plausible. Losing heaters and letdown due to low PZR level would greatly hamper the recovery effort, so this anwer is plausible.

Technical References: ES-0.4 Background Document

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPS04-3

Question Source:	Bank #	X	IPEC Bank 20796
	Modified Bank #		Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 5
	55.43	(b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	004000A213	
		Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Low RWST	
	Importance		3.9

Question # 86

A large break loss of coolant accident (LBLOCA) occurs. All equipment is available at the start of the event and functions as designed. In responding to this event, which of the following pumps could be secured first, and what is the procedural guidance for this action?

A. 21 RHR Pump in E-0, Reactor Trip or Safety Injection.

B. 22 Charging Pump prior to manipulating Recirc Switches in ES-1.3, Transfer to Cold Leg Recirculation.

C. 22 SI Pump from Recirc Switch 1 in ES-1.3, Transfer to Cold Leg Recirculation.

D. 21 and 22 RHR Pumps from Recirc Switch 3 Switches in ES-1.3, Transfer to Cold Leg Recirculation.

Answer: B

Explanation/Justification:

A. Incorrect. Plausible because for most accident conditions (not including LBLOCA) an RHR pump is secured first to prevent "Strong Pump - Weak Pump" interaction.
B. Correct. This action is performed to reduce loads on the 480V buses prior to transferring to recirculation.

C. Incorrect. Plausible because 22 SIP is secured first when Recirc Switch 1 is placed to ON; however, this action is performed after the charging pump is secured.

D. Incorrect. Plausible because 21 and 22 RHR Pumps are secured using Recirc Switch 3; however, this action is performed acter the charging pump is secured. Note: Recirc Switch 1 and 3 are placed to on in the same step.

Technical References: 2-ES-1.3

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPS13 – 4

Question Source: Bank # IPEC Bank

Modified Bank # Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge:

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b)

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	0080002237	
		Equipment Control - Ability to determine operability and/or availability of safety related equipment.	
	Importance		4.6

Question # 87

Unit 2 is cooling down in Mode 4 preparing for a refueling outage.

- 22 CCW was tagged out at the time of shutdown.
- 23 CCW trips while preparing to place RHR in service.

Which of the following is correct regarding the Tech Spec requirements for these conditions?

A. Two trains of CCW are inoperable. Since there is no AOT for this condition, LCO 3.0.3 applies.

B. One train of CCW is inoperable. Enter 72 hour AOT for this condition and continue cooldown.

C. Mode 5 cannot be entered because two trains of RHR will be required when Steam Generators can no longer be credited for RCS heat removal.

D. Only one CCW pump is required to accommodate normal and accident cooling loads, a Safety Function Determination can be performed to satisfy the CCW LCO.

Answer: B

Explanation/Justification:

A. Incorrect but plausible. One train of CCW is still operable based on these conditions. A candidate may not realize that only one train is inoperable.

B. Correct. This is not a "direct lookup" type of situation because with two inoperable pumps it is not a clear call, but a well prepared candidate should figure this out.

C. Incorrect but plausible. RHR cooling will be effected by this degraded CCW availability, so this answer is plausible. This is incorrect for a number of reasons. One reason is that the actual T.S. statement the plant is in requires going to Mode 5.

D. Incorrect but plausible. This is incorrect since there is no way to get relief from the CCW LCO via a safety function determination. This answer is plausible because the beginning of the statement is what T.S. basis says for CCW.

Technical References: Tech Specs 3.7.7

Proposed References to be provided: None

Learning Objective: I2LP-ILO-CCW001 – 13

Question Source: Bank # ______ IPEC Bank

Note changes or attach parent

Modified Bank # ______

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 4

55.43 (b) 2

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	0120002450	
		Emergency Procedures/Plan - Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	
	Importance		4.0

Question # 88

Given the following:

- The reactor is at 8% power preparing to synchronize the Main Generator to the grid.
- Bus 5 normal feed breaker tripped on overcurrent.
- The reactor remains critical.
- 21 & 24 RCPs trip on under voltage.
- 22 & 23 RCPs are operating.

Which of the following correctly describes the plant status and what if any actions should be taken?

A. No Reactor Protection Setpoints have been exceeded. Per the ARP, trip the Reactor and go to E-0, Reactor Trip or Safety Injection.

B. No Reactor Protection Setpoints have been exceeded. Per 2-AOP-138KV-1 Loss of Power to 6.9KV Bus 5 and/or 6, trip the Reactor and go to E-0, Reactor Trip or Safety Injection.

C. The reactor should have tripped on loss of flow in 2 loops. Per the ARP, trip the Reactor and go to E-0, Reactor Trip or Safety Injection.

D. Under frequency on 2 of 4 buses should have caused all RCPs to trip. Per 2-AOP-138KV-1 Loss of Power to 6.9KV Bus 5 and/or 6, trip the Reactor and go to E-0, Reactor Trip or Safety Injection.

Answer: A

Explanation/Justification:

A. Correct. 2 ARP-SAF directs a reactor trip if any pump is tripped regardless of power.
B. Incorrect because wrong procedure is given. Plausible it is reasonable that the AOP could specify tripping reactor
C. Incorrect. Plausible because the candidate may forget that the low flow trips (and alarms) are bypassed below P-7 (10% power).
D. Incorrect. Plausible because under frequency should trip all 4 RCPs; however, if power is lost, the under frequency will not trip the remaining RCPs.

Technical References: 2-ARP-SAF

Proposed References to be provided: None

Learning Objective: I2LP-ILO-RCSRCP - 15

Question Source: Bank # ______ IPEC Bank

Modified Bank # ______ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ______

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b)

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	059000A203	
		Ability to (a) predict the impacts of the following malfunctions or operations on the MFW System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Overfeeding event	
	Importance		3.1

Question # 89

Given the following:

- Unit 2 is operating at 100% when 21 MBFP trips to due to a Lovejoy malfunction.
- The automatic runback functions to lower turbine load, and SG NR Levels all lower to approximately 15% due to shrink.

Which of the following describes expected SG NR response and how this is addressed 2-AOP-FW-1, Loss of Feedwater?

A. Increase until a Turbine Trip occurs due to integral error in the feedwater control system. To prevent this, the AOP provides guidance to remove the integral error from the FRVs.

B. Stabilize below program level due to only having one MBFP in service. The AOP provides guidance to address this by controlling the MBFP and FRVs in manual as necessary to return level to program.

C. Increase to above program and then return to program level following a damped oscillation. The AOP does not provide guidance for any actions since level returns to program.

D. Increase and stabilize above program due to integral error signal induced by the transient. The AOP provides guidance to address this by controlling the MBFP and FRVs in manual as necessary to return level to program.

Answer: A

Explanation/Justification:
This question requires the candidate to know that the controllers for the FRVs are going to "windup" due to level being below program for an extended period of time. AOP-FW-1 would have the operators remove the “windup” from the FRVs, The candidate must remember this action and understand why it is done. With no actions, a Turbine trip will occur due to high level on SGs from overfeeding. Eliminating distractors will be easier if the candidate has a strong understanding of how the control system responds to being off program.

A. Correct.

B. Incorrect but plausible. The SGWLCS is level dominant, so it will only stabilize on program. This is plausible because there is one less feed pump, so level could be low.

C. Incorrect because without action, levels will increase until the Turbine trip occurs. Plausible because the system is going to try to do what this response describes.

D. Incorrect because of the level dominant SGWLCS. Plausible because there will be integral error from the transient.

Technical References:	2-AOP-FW-1
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-ICSGL – 5

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b)

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		90
	Group #		1
	K/A #	103000A203	
		Ability to (a) predict the impacts of the following malfunctions or operations on the Containment System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Phase A and B isolation	
	Importance		3.8

Question # 90

The following plant conditions exist on Unit 2:

- Reactor is at 100% RTP
- A manual Phase A isolation signal was inadvertently actuated on Train A.

Which of the following are direct results of this signal and corrective actions required to be taken in response to this event?

	Results	Actions
A.	LCV-459, Letdown Isolation Loop 21, and all orifice isolation valves close	Reset Phase A Restore Instrument Air to VC Place Excess Letdown in service
B.	LCV-459, Letdown Isolation Loop 21 and all orifice isolation valves close	Reset Phase A Restore Instrument Air to VC Place Letdown in service
C.	201, "Isolation Valve Letdown Line Normal Path" Isolation," and all orifice isolation valves close.	Reset Phase A Restore Instrument Air Place Letdown in service.
D.	201, "Isolation Valve Letdown Line Normal Path" Isolation" and all orifice isolation valves close.	Reset Phase A Restore Instrument Air to VC Place Excess Letdown in service

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because 459 does not isolate on a Phase A signal. There is no reason to place excess letdown in service if normal letdown is available.
B. Incorrect; Plausible because 459 does not isolate on a Phase A signal. The actions for this distractor are correct.
C. Correct. 201 and all orifice isolation valves do isolate on a Phase A signal. Since normal letdown is available, this would be preferred to excess letdown.
D. Incorrect. Plausible because the Results are correct but the actions are not. There is no reason to place excess letdown in service if normal letdown is available.

Technical References: 2-AOP-CVCS-1, 2-PT-R141

Proposed References to be provided: None

Learning Objective: I2LP-ILO-CVCS – 5

Question Source: Bank # ____ IPEC Bank

Modified Bank # ____ Note changes or attach parent

New X

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge: ____

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b)

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	045000A217 Ability to (a) predict the impacts of the following malfunctions or operations on the MT/G System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Malfunction of electrohydraulic control	
	Importance		2.9

Question # 91

The plant is at 50% power. A malfunction has occurred resulting in an 8 psig increase in Main Turbine Governor oil pressure. Which of the following correctly identifies the impact of this malfunction and the required operator actions as specified in 2-AOP-LOAD-1, Excessive Load Increase or Decrease?

	Effect of Malfunction	Required Actions
A.	• Load will increase • Load increase will be limited by the aux governor	If two PR NIs are ≥ 108% then trip the reactor and go to E-0, Reactor trip or Safety Injection
B.	• Load will increase • Load increase will be limited by the Load Limit	Adjust Turbine load to restore Tavg to within 1.5°F of Tref Restore ΔI to the program band
C.	• Load will increase • Load increase will be limited by the Load Limit	Withdraw control rods to restore Tavg to within 1.5°F of Tref and restore ΔI to the program band
D.	• Load will increase • Load increase will not be limited	If two PR NIs are ≥ 108% then trip the reactor and go to E-0, Reactor trip or Safety Injection

Answer: B

Explanation/Justification:

Governor Oil pressure increases from approximately 20 psig to approximately 40 psig from Latched to full power. At 50% power the pressure is approximately 28 psig. When power is < 75% the load limits are maintained within 8 psig of governor oil pressure. When the governor oil pressure increases above Load Limit oil pressure, the Load Limit oil pressure will be in control. The AOP directs turbine load reduction to restore Tavg not rod adjustment.

A. Incorrect. Plausible because the aux governor will limit load if it is increasing at 3% per second. If 2 PR NIs exceed 108% the reactor should be tripped; however a 8 psig (Maximum increase) from 50% will not cause power to exceed 108%

B. Correct

C. Incorrect. Plausible because the load will be limited by the load limit setpoint; however, restoring Tavg using control rods is not desired and may make ΔI worse without a boron adjustment.

D. Incorrect. Plausible because the candidate must remember that the load limit oil pressures are maintained within 8 psig of control oil pressure. Furthermore the candidate must remember that the lower of the oil pressures (governor or load limit) controls the turbine. If 2 PR NIs exceed 108% the reactor should be tripped; however an 8 psig (Maximum increase) from 50% will not cause power to exceed 108%.

Technical References:		2-AOP-LOAD-1	
Proposed References to be provided:		None	
Learning Objective:		I2LP-ILO-MTG001 – 7 I2LP-ILO-MTG001 – 5	
Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	
Question History:	Last 2 NRC Exams at IPEC:		NA
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:		X
10 CFR Part 55 Content:	55.41	(b)	
	55.43	(b) 5	
Comments:			

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	0620002240	
		Equipment Control - Ability to apply technical specifications for a system.	
	Importance		4.7

Question # 92

The normal supply breaker to 480V Bus 3A opens due to a breaker malfunction.

The crew responds per 2-AOP-480V-1, Loss of Normal Power to Any 480V Bus, and is unable to close breaker EG-2B.

There is no damage to 3A and all fault indications are cleared.

Based on these conditions, what actions will be directed by 2-AOP-480V-1 to ensure compliance with Technical Specifications?

A. Regardless of mode, no actions will be taken to power any of the loads normally fed from 480V Bus 3A until the bus can be powered from either its normal feed or 22 EDG.

B. Rack in and close breaker 2AT3A if RCS temperature is < 200^0F since there is no damage to Bus 3A.

C. Rack in and close breaker 2AT3A if RCS temperature is < 350^0F since there is no damage to Bus 3A.

D. Regardless of mode, rack in and close breaker 2AT3A since there is no damage to Bus 3A and enter AOT for 480V Safeguards Busses 2A/3A inoperable.

Answer: B

Explanation/Justification:

The T.S. Basis 3.8.2 page 3 at the bottom of the page spells out that 480V busses cannot be tied above 200F.

A. Incorrect but plausible. The procedure will have Unit 1 power backup supplied to 23 Battery Charger. Also 2AT3A can be closed if temperature is <200F. It is plausible because the AOP may not ever specifify tying the breakers.

B. Correct. Based on 2-AOP-480V-1 Attachment 2 step 2.160/2.161

C. Incorrect but plausible. The mode makes this selection incorrect. It is plausible because many safeguards requirements are relaxed below 350F.

D. Incorrect but plausible. This is plausible because closing the tie breaker would only jeopardize the 2A and 3A busses. Since the answer states that these busses would be declared inoperable, it is plausible that the procedure could specify this.

Technical References:	2-AOP-480V-1 Att 2 Tech Specs 3.8.2 Basis
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-AOP480 - 2

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	014000A202	
		Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: - Loss of power to the RPIS	
	Importance		3.6

Question # 93

The unit is operating at 100% power.

The following annunciators are in alarm:

- Approaching Rod Insertion Limit 1.25"
- Rod Insertion Limit 0"
- Rod Control Non Urgent Failure
- Rod Bottom Rod Stop

All IRPIs indicate 0" and all rod bottom lights are extinguished.

What event has occurred and what actions are required?

	EVENT	ACTIONS
A.	MCC-24 is de-energized	Be in MODE 3 in 6 hours
B.	MCC-24 is de-energized	Place control rods under manual control
C.	23 Instrument Bus is de-energized	Reduce THERMAL POWER to ≤ 75% RTP
D.	23 Instrument Bus is de-energized	Verify SDM to be within the limits specified in the COLR

Answer: B

Explanation/Justification:

A. Incorrect. Plausible because MCC-24 is the power supply to the IRPI. TS action is not correct.

B. Correct. MCC-24 is the power supply to the IRPI. All alarms and indications are consistent with a loss of power. Placing rod control in manual satisfies TS 3.1.7.

C. Incorrect. Plausible because 23 Instrument bus supplies most of the indications and controllers on the flight panel. Reducing thermal power to <75% is a TS action for a misaligned rod.

D. Incorrect. Plausible because 23 Instrument bus supplies most of the indications and controllers on the flight panel. Verification of SDM is TS action for misaligned/dropped rods.

Technical References:	Tech Spec 3.1.7 2-AOP-480V-1
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-ICRPI – 7 I2LP-ILO-ICRPI – 14

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b)
	55.43	(b) 2

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		Generic
	Group #		Conduct of Ops
	K/A #	1940012107	
		Conduct of Operations - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	
	Importance		4.7

Question # 94

While determining if RCPs should be stopped in ECA-2.1, Uncontrolled Depressurization of All Steam Generators, the following plant conditions exist:

- Reactor Coolant System Pressure is 1035 psig and stable.
- Hot Leg temperature ~ 340ºF and stable.
- Cold Leg temperature ~ 325ºF and stable.
- Subcooling ~ 200ºF.
- All Reactor Coolant Pumps running.
- Steam Generator Pressures

21	22	23	24
250 psig increasing	230 psig decreasing	230 psig decreasing	250 psig stable

Which ONE of the following is the correct course of action?

A. Transition to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

B. Continue in ECA-2.1, Uncontrolled Depressurization of All Steam Generators

C. Transition to E-2, Faulted Steam Generator Isolation

D. Transition to E-3, Steam Generator Tube Rupture

Answer: C

Explanation/Justification:

A. Incorrect but plausible since FR-P.1 criteria are almost met.
B. Incorrect but plausible since there are situations where transition to E2 is delayed when a MSIV is closed.
C. Correct based on 2-ECA-2.1 foldout Page.
D. Incorrect but plausible since an operator could mistake the increasing SG pressure as being due to a SGTR.

Technical References: 2-ECA-2.1 Foldout Page

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPC21 – 1

Question Source: Bank # X IPEC Bank 15864

Modified Bank # Note changes or attach parent

New

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge:

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 5

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		Generic
	Group #		Conduct of Ops
	K/A #	1940012145	
		Conduct of Operations - Ability to identify and interpret diverse indications to validate the response of another indication.	
	Importance		4.3

Question # 95

Given:

- Unit 2 has experienced a large break LOCA.
- All safeguards equipment operated as designed.
- The crew has transitioned to 2-ES-1.3.
- ONE RWST Low Low Level alarm is illuminated.

Which of the following is used to determine if a level transmitter has failed?

A. Compare RWST level to Containment level

B. Check time from SI initiation > 30 minutes

C. Check Containment Sump level increasing

D. Check Containment Sump level > 46' 8 ½ "

Answer: D

Explanation/Justification:

A. Incorrect. Plausible because ECA-3.1 (Not ES-1.3) has a graph to compare RWST level with expected Containment level.

B. Incorrect. Plausible because the length of time to reach the low low level setpoint is approximately 25 minutes

C. Incorrect. Plausible because checking the level increasing is done if both RWST low low level alarms are illuminated; however to confirm adequate level for recirc/RHR pump operation requires checking containment sump level (i.e., actual sump level may be inadequate for pump NPSH).

D. Correct. When RWST is at approximately 9.24', containment level should be approximately 46' 9 ½ " to provide adequate NPSH for the recirc/RHR pumps.

Technical References:	2-ES-1.3 Background
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-EOPS13 – 4

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 7
	55.43	(b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		Generic
	Group #		Equip Control
	K/A #	1940012222	
		Equipment Control - Knowledge of limiting conditions for operations and safety limits.	
	Importance		4.7

Question # 96

While operating at 30% power, rod control problems result in ΔFlux being outside of the target band from 0100 until 0330. Which of the following statements describes the limitations that will be applied to bring the reactor to 100% power? Assume that the ΔFlux was restored at 0330 and no further problems were encountered.

A. Power may be increased above 50% as soon as ΔFlux is in the target band. ΔFlux limits do not apply below 50% power.

B. Power may not be raised above 50% until 0130 the next day.

C. Power may not be raised above 50% until 0230 the next day.

D. Power may be increased above 50% as soon as the ΔFlux is in the target band. You can accumulate up to 16 hours of time outside of the target band without penalty.

Answer: B

Explanation/Justification:

The total time ΔFlux is out of the target band is 2.5 hours. Penalty minutes accumulate at ½ minute per minute out of the band, when < 50% power, for a total of 1 hour and 15 penalty minutes.

When Greater than 60 penalty minutes are obtained. Power cannot be raised above 50% until these minutes roll off to <60. These minutes roll off at the rate they were accumulated. Because of this, the 15 minutes of penalty time needed to go above 50% will require 30 min of operation 24 hours after delta flux originally went out of the band. Therefore at 0130 the next day power can be raised above 50%.

A. Incorrect. Plausible because part of the statement is correct. T.S. says AFD “may deviate outside the target band with thermal power <50%”.

B. Correct. While penalty minutes accumulate at ½ minute per minute out of the band. At 0130 the minutes outside the target are < 60 in the previous 24.

C. Incorrect. Power can be raised above 50% at 0130. Candidate must properly calculate the time that the penalty minutes “roll off”.

D. Incorrect. This is plausible because there is a note in the T.S. allowing a total of 16 hours outside the band is allowed without penalty deviation during surveillance of power range channels.

Technical References:	Tech Spec 3.2
Proposed References to be provided:	None
Learning Objective:	I2LP-ILO-ICROD – 14

Question Source:	Bank #	X	IPEC Bank 8436
			Note changes or
	Modified Bank #		attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis:	

10 CFR Part 55 Content:	55.41	(b) 5
	55.43	(b) 2

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		Generic
	Group #		Equip Control
	K/A #	1940012221	
		Equipment Control - Knowledge of pre- and post-maintenance operability requirements.	
	Importance		4.1

Question # 97

Given the following conditions:

- Maintenance requested a tagout for 23 SI Pump
- The tagout included placing the control room switch in pullout and racking out the supply breaker only
- A visual inspection showed no work was required
- No disassembly work was performed

Which ONE of the following indicates the minimum requirement for restoring 23 SI Pump operability?

23 SI Pump can be considered operable when:

A. Its supply breaker is racked in and its control room switch is back in AUTO

B. Its supply breaker is racked in and its control room switch is back in AUTO and the pump has been started using the control room switch.

C. Its supply breaker is racked in and its control room switch is back in AUTO and the pump has been started using the control room switch and verified to meet the acceptance criteria of the quarterly surveillance test.

D. Its supply breaker is racked in and its control room switch is back in AUTO and the pump has been started using the control room switch AND using the auto-start relay.

Answer: B

Explanation/Justification:

A. Incorrect but plausible. A candidate may believe that once the breaker is racked in that the pump would auto start.

B. Correct. The minimum requirement is to have the equipment in the proper configuration and verify the breaker will close and start the pump

C. Incorrect but plausible. It is not necessary to verify the pump develops head etc. because nothing that was done calls in to question that this was affected. This is plausible because we generally schedule the routine surveillance to occur after removing a PTO for maintenance

D. Incorrect but plausible. It is not necessary to verify that the pump auto start circuitry works. A candidate could conclude it is because racking out the breaker may affect breaker cell switches requiring testing auto-start.

Technical References:	OAP-37		
Proposed References to be provided:	None		
Learning Objective:	I0LP-ILO-ADM01 – 1		
Question Source:	Bank #		IPEC Bank
	Modified Bank #	Kewaune e 2002	Note changes or attach parent
	New		
Question History:	Last 2 NRC Exams at IPEC:		NA
Question Cognitive Level:	Memory or Fundamental Knowledge:		
	Comprehension or Analysis:		X
10 CFR Part 55 Content:	55.41	(b)	
	55.43	(b) 2	

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	00WE13234	
		Radiological Controls - Knowledge of radiation exposure limits under normal and emergency conditions.	
	Importance		3.7

Question # 98

Given the following:

- A severe accident has occurred at Unit 2. 22 SG has extremely high activity levels due to the accident.
- An over pressure condition exists on 22 SG that threatens to lift a safety valve.
- Local operation of 22 SG Atmospheric Dump Valve (ADV) is required to lower SG pressure in a controlled manner.
- The dose rate at the local ADV controls is 20 Rem/hr and the evolution is expected to take 30-45 minutes.
- A Reactor Operator who has not entered the RCA this year has volunteered to perform the task.

Which of the following statements is correct regarding this evolution?

A. The evolution cannot be performed because the individual is expected to exceed their NRC occupational limit.

B. The evolution may be performed. The operator must be a volunteer and the exposure must be approved by any active SRO.

C. The evolution may be performed. The operator must be a volunteer and the exposure must be approved by the EPM/POM or Emergency Director.

D. The evolution may be performed. The operator does NOT have to be a volunteer since the expected exposure is < 25 Rem. The exposure must be approved by EPM/POM or Emergency Director.

Answer: C

Explanation/Justification:

A. Incorrect but plausible because the NRC normal limit will be exceeded.
B. Incorrect but plausible. Any active SRO is not authorized to approve exposure.
C. Correct
D. Incorrect but plausible. The individual has to be a volunteer.

Technical References:	EP Form 6
Proposed References to be provided:	None
Learning Objective:	I0LP-ILO-ERT003 - 3

Question Source:	Bank #		IPEC Bank
	Modified Bank #		Note changes or attach parent
	New	X	

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b)
	55.43	(b) 7

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		Generic
	Group #		Emerg Proc/Plan
	K/A #	1940012404	
		Emergency Procedures/Plan - Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	
	Importance		4.7

Question # 99

The following plant conditions exist:

- The plant was operating at 100% power
- The Team has initiated a manual reactor trip and Safety Injection
- RCS pressure is 1650 psig and slowly decreasing
- SG pressures are all stable at 900 psig
- No SG Level is increasing in an uncontrolled manner
- Auxiliary feedwater flow is 450 gpm and stable
- All SI pumps are running
- Containment Radiation levels, CNMT Temperature and pressure remain normal
- Primary Auxiliary Building radiation levels are increasing
- All secondary side radiation monitor readings are normal

If the RCS leakage cannot be isolated, which ONE of the following EOP procedure sequences would be utilized to address these conditions upon transition from E-0 Reactor Trip or Safety Injection?

E-1, Loss of Reactor or Secondary Coolant
ECA-1.2, LOCA Outside of Containment
ECA-1.1, loss of Emergency Coolant Recirculation

A. E-1 to ECA-1.2 to E-1.

B. ECA-1.2 to ECA-1.1.

C. E-1 to ECA-1.2 to ECA-1.1.

D. ECA-1.2 to E-1.

Answer: B

Explanation/Justification:

A. Incorrect but plausible because an operator may not know that E-1 is not entered prior to ECA-1.2 and that E-1 is not where the team goes upon exit from ECA-1.2 if break is not isolated.

B. Correct. The procedure flowpath would be to go through E-0 and not go to E-1 at step 15 because conditions inside containment will be normal. At step 22 of E-0, there will be an RNO step to go to ECA-1.2. At step 6 of ECA-1.2 pressure will not be increasing so transition to ECA-1.1 will be made.

C. Incorrect but plausible because may not know that E-1 is not entered prior to ECA-1.2.

D. Incorrect but plausible because an operator may not know that E-1 is not where the team goes upon exit from ECA-1.2.

Technical References: 2-E-0, 2-ECA-1.2

Proposed References to be provided: None

Learning Objective: I2LP-ILO-EOPC12 – 5

Question Source: Bank # X IPEC Bank 8315

Modified Bank # Note changes or attach parent

New

Question History: Last 2 NRC Exams at IPEC: NA

Question Cognitive Level: Memory or Fundamental Knowledge:

Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (b) 10

55.43 (b) 5

Comments:

Exam Outline Cross Reference:	Level	RO	SRO
	Tier #		Generic
	Group #		Emerg Proc/Plan
	K/A #	1940012430	
		Emergency Procedures/Plan - Knowledge of which events related to system operations/status that must be reported to internal orginazations or external agencies, such as State, the NRC, or the transmission system operator.	
	Importance		4.1

Question # 100

The plant is in an outage in Mode 5 when a complete loss of RHR occurs. Temperature increases and is stabilized at 220^0F by the SG Atmospheric Dump Valves throttling.

Which ONE of the following identifies when the NRC is required to be notified of this event?

Notify the NRC within...

A. 1 hour

B. 4 hours

C. 8 hours

D. 30 days

Answer: A

Explanation/Justification:
EAL 8.2.3 met. E-Plan declaration requires 1 hour report to NRC

A. Correct
B. Incorrect but plausible. The other available report times are all valid for various equipment failures.

C. Incorrect but plausible. The other available report times are all valid for various equipment failures.

D. Incorrect but plausible. The other available report times are all valid for various equipment failures.

Technical References:	IP-SMM-LI-108
Proposed References to be provided:	None
Learning Objective:	I0LP-ILO-ADM01 – 1

Question Source:	Bank #		IPEC Bank
	Modified Bank #	Nine Mile Point 2002	Note changes or attach parent
	New		

Question History:	Last 2 NRC Exams at IPEC:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge:	
	Comprehension or Analysis:	X

10 CFR Part 55 Content:	55.41	(b)
	55.43	(b) 5

Comments: