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EXAMPLE OF STUDENT SCORE:

PART 1

*Rtype: Z01.47(Ind) / Z01.48 (Key)
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TRAINING DEPARTMENT EXAMINATION

1. *NAME: KEY
   Last, Initials, (Cross Ref.)

2. *logon/KCN: KEY
   (Cross Ref.)

3. *ONTRACK: **RWEX-NRC-MAR-14RO**
   (Cross Ref.)

4. *Exam Date: 14-03-21
   (Doc Date) YY-MM-DD

5. *EXAM NO.: **RWEX-NRC-MAR-14RO**
   (Document No.)

6. **EXAM TITLE: 2014 March RO NRC Exam**
   (Doc. Title)

7. EXAM GRADE:________ 8. GRADED BY________________ 9. TOTAL PTS: 75

I have read and understand the rules and guidelines on the Written Examination Coversheet, Form ES-401-7. In addition, to the best of my knowledge, I did not compromise the integrity of the examination process and will immediately report to management any indications or suggestions that examination security may have been compromised.

KEY

10. KEY
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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

(T) (F) KEY

% 2 3 5

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

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

TO USE SUBJECTIVE SCORE FEATURE:

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EXAMPLE OF STUDENT SCORE:

PART 2

NAME			
SUBJECT		TEST NO.	
DATE		PERIOD	

TEST RECORD	
PART 1	
PART 2	
TOTAL	

KEY

	(T)	(F)			KEY
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52	●	B	C	D	E
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97	A	B	C	D	E
98	A	B	C	D	E
99	A	B	C	D	E
100	A	B	C	D	E

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*Rtype: Z01.47(Ind) / Z01.48 (Key)
Circle applicable record type

KEY **TRAINING DEPARTMENT EXAMINATION**

1. *NAME: KEY
Last, Initials, (Cross Ref.)

2. *logon/KCN: KEY
(Cross Ref.)

3. *ONTRACK: **RWEX-NRC-MAR-14SRO**
(Cross Ref.)

4. *Exam Date: 14-03-21
(Doc Date) YY-MM-DD

5. *EXAM NO.: **RWEX- NRC-MAR-14SRO**
(Document No.)

6. **EXAM TITLE: 2014 March SRO NRC Exam**
(Doc. Title)

7. EXAM GRADE:________ 8. GRADED BY________________ 9. TOTAL PTS: 100

I have read and understand the rules and guidelines on the Written Examination Coversheet, Form ES-401-7. In addition, to the best of my knowledge, I did not compromise the integrity of the examination process and will immediately report to management any indications or suggestions that examination security may have been compromised.

KEY

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PART 1

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

(T) (F) KEY

	Key
1	A
2	B
3	B
4	A
5	C
6	D
7	A
8	C
9	D
10	B
11	C
12	D
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34	B
35	D
36	A
37	A
38	B
39	D
40	C
41	B
42	D
43	C
44	A
45	A
46	B
47	A
48	B
49	B
50	D

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- 153 points maximum

EXAMPLE OF STUDENT SCORE:

NAME			
SUBJECT		TEST NO.	
DATE		PERIOD	

TEST RECORD	
PART 1	
PART 2	
TOTAL	

PART 2

KEY

	(T)	(F)			KEY
	%	2	3	■	5
51	A	B	■	D	E
52	■	B	C	D	E
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54	A	B	■	D	E
55	■	B	C	D	E
56	A	■	C	D	E
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97	A	B	C	■	E
98	A	B	■	D	E
99	A	■	C	D	E
100	A	B	C	■	E

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QUESTION 1 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295001 AA1.05		IR 3.3

Ability to operate and/or monitor the following as they apply to Partial or Complete Loss of Forced Core Flow Circulation:
AA1.05 Recirculation flow control system

Proposed Question:

Due to a failure of Reactor Recirculation Pump "B" an unexpected entry into the Restricted Region of the Power to Flow Map has occurred. The CRS has directed immediate exit of the Restricted Region using the Recirculation Flow Control System.

To comply with this direction, the ATC operator should toggle B33-K603A in the ___(1)___ direction using ___(2)___ detent.

A. (1) open; (2) slow

B. (1) open; (2) fast

C. (1) close; (2) slow

D. (1) close; (2) fast

Proposed Answer: A.

Explanation:
A. Correct- Flow should be raise to exit the region. Lowering flow would result in further instability. Despite the word immediate, the FCVs are operated in slow detent to maintain control of the evolution.
B. Part 1 is correct, but the FCVs are operated in slow detent to maintain control of the evolution.
C. Toggling to the close direction would result in further instability. The region must be exited by raising flow. Part 2 is correct.
D. Toggling to the close direction would result in further instability. The region must be exited by raising flow. Despite the word immediate, the FCVs are operated in slow detent to maintain control of the evolution.

Technical Reference(s): SOP-0003 Rev 311 Pg 37 of 100; AOP-0024 Rev 25 Pg 8 of 16

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053 Obj 1j, 7e & RLP-OPS-AOP024 Obj 5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 2 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295003 G.2.1.20		IR 4.6

Ability to interpret and execute procedure steps during a partial or complete loss of AC power.

Proposed Question:

A loss of offsite power has occurred. All 3 diesel generators have started and are supplying associated buses.

Which of the following lists the immediate operator actions for the applicable Abnormal Operation Procedure (AOP)?

A. Manually initiate Reactor Core Isolation Cooling (RCIC). Bypassing of interlocks is only allowed by the Emergency Operating Procedures.

B. Manually initiate RCIC AND bypass high temperature isolation interlocks.

C. Manually initiate High Pressure Core Spray (HPCS). Bypassing of interlocks is only allowed by the Emergency Operating Procedures.

D. Manually initiate HPCS, AND defeat the high water level isolation interlock.

Proposed Answer: B.

Explanation:
A. Initiation of RCIC is correct, however the AOP also directs bypassing high temperature interlocks.
B. Correct – AOP-0004, Loss of Offsite Power directs Initiation of RCIC and bypassing of high temperature isolation interlocks.
C. Although the availability of the Div 3 DG keeps HPCS as an available injection source, the AOP directs the initiation of RCIC and bypassing of high temperature interlocks.
D. Although the availability of the Div 3 DG keeps HPCS as an available injection source, the AOP directs the initiation of RCIC and bypassing of high temperature interlocks. The bypassing of the high water level isolation interlock is only authorized by the Emergency Operating Procedures.

Technical Reference(s): AOP-0004, Rev 50, Pg 4 of 72.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0523 Obj 2, 5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.10

Comments: Although the recall of immediate operator actions would normally be memory or fundamental knowledge, the cognitive level of this question is comprehension or analysis since the examinee must evaluate the conditions to determine which AOP is applicable.

QUESTION 3 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	1	Group # 1
	K/A #	295004 AA1.02	IR 3.8

Ability to operate and/or monitor the following as they apply to Partial or Complete Loss of D.C. Power:
AA1.02 Systems necessary to assure safe plant shutdown

Proposed Question:

The plant is recovering from a loss of coolant accident. Residual Heat Removal (RHR) pump "A" is running in the Suppression Pool Cooling mode when the pump breaker experiences a loss of DC control power.

RHR "A" pump will _____________.

A. trip and all indicating lights will be extinguished.

B. remain running with all indicating lights extinguished.

C. trip but only the white indicating light will be extinguished.

D. remain running with only the white indicating light extinguished.

Proposed Answer: B.

Explanation:
A. The pump will not trip on loss of DC control power. The pump must be manually tripped locally.
B. Correct – The pump will remain running without trip capability and all indicating lights will lose power.
C. The pump will not trip, and all lights will be extinguished.
D. The pump will remain running, but all indicating lights will be extinguished.

Technical Reference(s): R-STM-0305 Rev 6, Pg 13 of 37; AOP-0014 Rev 23 Pg 19 of 74

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0605 Obj 10b

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:
Removed the word "all" from distractor C

QUESTION 4 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	1	Group # 1		
	K/A #	295005 AK2.08			IR 3.2

Knowledge of the interrelations between Main Turbine Generator Trip and the following:
AK2.08 A.C. electrical distribution

Proposed Question:

Following an automatic Turbine-Generator Trip from 100% power, a ___(1)___ transfer will occur resulting in NPS-SWG1A being supplied from ___(2)___.
(Assume all systems work as designed)

A. (1) fast; (2) RTX-XSR1E, Preferred Station Transformer

B. (1) slow; (2) RTX-XSR1E, Preferred Station Transformer

C. (1) fast; (2) STX-XNS1A, Normal Station Transformer

D. (1) slow; (2) STX-XNS1A, Normal Station Transformer

Proposed Answer: A.

Explanation:

A. Correct -With all systems working as designed as stated in the stem, a FAST transfer will occur allowing a transfer of power while maintaining continuity without a loss of loads. During the transfer, the loads transfer from the Normal Station Transformer which were supplied by the Main Generator to the Preferred Station Transformers which are supplied by an offsite power source. Had this been a manual trip of the output breakers, a slow transfer would have occurred.

B. A slow transfer only occurs if the fast transfer is unsuccessful (or if the output breakers are manually opened). Part 2 is correct.

C. Part 1 is correct, but Part 2 is incorrect because the Normal Station Transformers will not longer have power when the Main Generator trips.

D. A slow transfer only occurs if the fast transfer is unsuccessful (or if the output breakers are manually opened) and the Normal Station Transformers will not longer have power when the Main Generator trips.

Technical Reference(s): R-STM-0300 Rev 26 Pg 16-17 of 105,

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0300 Obj 3d, 14c, 18a

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7

Comments: Original question used NNS-SWG1A; changed to NPS and changed transformers to reflect NPS from XSR1C to E and XNS1C to A

QUESTION 5 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	1	Group # 1		
	K/A #	295006 AK1.03			IR 3.7

Knowledge of the operational implications of the following concepts as they apply to SCRAM:
AK1.03 Reactivity control

Proposed Question:

To ensure sufficient time is allowed for full control rod insertion, the scram signal from the ___(1)___ is present ___(2)___ to prevent premature scram reset.

A. (1) RPS manual pushbuttons; (2) for 10 seconds

B. (1) RPS manual pushbuttons; (2) until bypassed with a keylock switch

C. (1) Mode Switch; (2) for 10 seconds

D. (1) Mode Switch; (2) until bypassed with a keylock switch

Proposed Answer: C.

Explanation:
A. The signal which ensure full insertion of control rods during a scram is present for 10 seconds after placing the mode switch in shutdown; not the RPS manual pushbuttons.
B. The signal which ensure full insertion of control rods during a scram is only present for 10 seconds after placing the mode switch in shutdown; not the RPS manual pushbuttons.
C. Correct – The reactor mode switch in shutdown provides a redundant scram signal. After 10 seconds, this signal is automatically bypassed to allow resetting the scram. During the initial 10 seconds, the scram can not be reset ensuring adequate shutdown margin by allowing all control rods to fully insert.
D. Part 1 is correct, but the signal is only present for 10 seconds.

Technical Reference(s): R-STM-0508 Rev 6 Pg 15 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508 Obj 5c

Question Source: Bank # December 2010 NRC Exam #5

Question History: Last NRC Exam December 2010 NRC Exam #5

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.6

Comments: Appeared on one of last 2 NRC exams (1 of 3)

QUESTION 6 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295016 G2.4.34		IR 4.2

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects related to Control Room Abandonment.

Proposed Question:

Due to a fire in the Main Control Room, transfer of control has been established at the Division 1 Remote Shutdown Panel in accordance with AOP-0031, SHUTDOWN FROM OUTSIDE THE MAIN CONTROL ROOM.

Following the transfer which modes of Safety Relief Valve operation will still be functional?

A. Automatic Depressurization and Relief

B. Manual and Automatic Depressurization

C. Relief and Safety

D. Safety and Manual

Proposed Answer: D.

Explanation:
A. Once control has been transferred to the RSS panel, ADS and Relief will no longer function.
B. Manual is correct, but once control is transferred to the RSS panel, ADS will no longer function.
C. Safety is correct, but once control is transferred to the RSS panel, Relief will no longer function.
D. Correct – After control is transferred, the SRVs will still operate in the Safety mode and the valves will still be able to be open in Manual. During a fire, power to ENB-PNL02A(B) is de-energized but once transfer of control is complete the SRVs receive power from ENB-PNL03A(B).

Technical Reference(s): AOP-0031 Rev 322 Pg 27 of 122

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0200 Obj 6

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 7 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295018 AA2.03		IR 3.2

Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Component Cooling Water:
AA2.03 Cause for partial or complete loss

Proposed Question:

The following plant conditions exist:

• Reactor water level	-10 inches
• Drywell Pressure	1.92 psid
• CRD Charging Water Header Pressure	1680 psig

The following annunciator windows are illuminated:

• F/D INLET HIGH TEMP 130 DEG F	(H13-P680/01A/C01)
• RECIRC PUMP A SEAL CLG WATER LOW FLOW	(H13-P680/04A/C05)
• RECIRC PUMP A WINDING CLG WATER LOW FLOW	(H13-P680/04A/C06)
• RECIRC PUMP B SEAL CLG WATER LOW FLOW	(H13-P680/04A/C11)
• RECIRC PUMP B WINDING CLG WATER LOW FLOW	(H13-P680/04A/C12)

The cause for the above indications is______________.

A. High drywell pressure

B. 50 psig CCP header pressure

C. Low reactor water level

D. Manual initiation of Standby Service Water

Proposed Answer: A.

Explanation:
A. Correct - High drywell pressure isolates CCP to containment resulting in loss of cooling to the RWCU heat exchanger and Recirc Pump seal and winding coolers.
B. If pressure were this low, the CRD pumps would trip and the indicated charging water header pressure would be 0 psig.
C. The CCP isolation valves would not close until reactor water level falls to level 2 (-43").
D. Manual initiation of SSW would result in a trip of the CRD pump therefore charging water header pressure would be 0 psig.

Technical Reference(s): AOP-0003 Rev 33, Pg 15 of 22
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-OPS-AOP11 Rev 3 Obj 2
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3
10 CFR Part 55 Content: 55.41.b.7
Comments:

QUESTION 8 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295019 AK2.02		IR 2.9

Knowledge of the interrelations between Partial or Complete Loss of Instrument Air and the following:
AK2.02 Component cooling water

Proposed Question:

During a loss of instrument air, MWS-AOV134, CCP SURGE TANK MAKEUP VALVE fails ___(1)___ and the CCP-TVX128, HEAT EXCHANGER BYPASS VALVE fails ___(2)___.

A. (1) open; (2) open

B. (1) open; (2) closed

C. (1) closed; (2) closed

D. (1) closed; (2) open

Proposed Answer: C.

Explanation:
A. MWS-AOV134 will fail closed and CCP-TVX128 will also fail closed.
B. MWS-AOV134 will fail closed. Part 2 is correct.
C. Correct – Both valves will fail closed on loss of air.
D. Part 1 is correct. CCP-TVX128 will fail closed to force flow through the heat exchangers for maximum cooling.

Technical Reference(s): PID-09-01A, PID-09-01B and PID-09-15B

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115 Obj 3d, 4a, 5e

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 9 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	1	Group #	1	
	K/A #	295021 AK3.05			IR 3.6

Knowledge of the reasons for the following responses as they apply to Loss of Shutdown Cooling:

AK3.05 Establishing alternate heat removal flowpaths

Proposed Question:

The plant has been in a refueling outage for 10 days when the operating shutdown cooling pump trips. Plant procedures require establishing an alternate heat removal flowpath in order to ________________________.

A. remove sensible heat present due to extended plant operation

B. provide flow to nuclear instrumentation in the core.

C. avoid vessel chemistry concerns due to stagnation from low flow conditions

D. remove decay heat generated by the decay of fission products

Proposed Answer: D.

Explanation:

A. Although a concern in the firt 48 hours after shutdown, at 10 days into the outage, decay heat is the main concern.

B. Although temperature monitoring is important, removal of decay heat is the prime concern.

C. Loss of SDC does not affect flow through RWCU therefore this would not cause a concern regarding vessel chemistry.

D. Correct - Decay heat removal capability must be maintained due to the continuous generation of heat caused by the heat of fission products. Insufficient removal of heat will result in overheating of the core and clad damage.

Technical Reference(s): AOP-00051 Rev 312 Pg 3 of 30

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0543 Obj 1

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.14

Comments:
Revised distractor B to read “provide flow to nuclear instrumentation in the core”.

QUESTION 10 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295023 AK1.03		IR 3.7

Knowledge of the operational implications of the following concepts as they apply to Refueling Accidents:
AK1.03 Inadvertent criticality

Proposed Question:

The bases of the refueling position one-rod out-interlock is to:

A. Prevent possible fuel damage caused by withdrawing a control rod from a fuel cell containing less than four fuel assembles.

B. Prevent the reactor from becoming critical during refueling operations.

C. Prevent the potential for draining the cavity / vessel during the performance of CRDM maintenance.

D. Prevent the movement of the refueling bridge over the reactor core with a control rod withdrawn.

Proposed Answer: B.

Explanation:
A. Administrative controls are utilized to prevent the possibility of fuel damage in fuel cells containing less than four fuel assembles
B. Correct – Technical Specification 3.9.2 bases background states that the refuel position one-rod-out interlock restricts the movement of control rods to reinforce unit procedures that prevent the reactor from becoming critical during refueling operations.
C. The one-rod-out interlock prevents withdrawal of more than one control rod and does not provide protection from an OPDRV during the performance of CRDM maintenance.
D. This is a Refuel Equipment Interlock and is governed under Technical Specification 3.9.1.

Technical Reference(s): RBS Technical Specification 3.9.2

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0055 Obj. 9

Question Source: New;

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41b.10

Comments:
Replaced Question

QUESTION 11 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 295024 EK3.03		IR 3.6

Knowledge of the reasons for the following responses as they apply to High Drywell Pressure:
EK3.03 Containment venting: Mark III

Proposed Question:

A leak in the drywell has resulted in elevated pressure in the primary containment.

In accordance with EOP-0002, PRIMARY CONTAINMENT CONTROL, containment venting is required BEFORE containment pressure reaches 30 psig with allowance to exceed release rate limits in order to prevent _________________________.

A. Standby Gas Filter Train damage due to exceeding design inlet pressure.

B. SRV tailpipe damage due to exceeding the code allowable stresses

C. containment damage resulting in a large uncontrolled release of radioactivity due to the inability to open ventilation dampers above this pressure

D. containment damage resulting in a large uncontrolled release of radioactivity due to exceeding the pressure suppression pressure

Proposed Answer: C.

Explanation:

A. This containment pressure leg of EOP-0002 has guidance to avoid damage to the AB ductwork, but the direction is to stop venting when containment pressure can not be maintained below 2 psig.

B. The SRV tailpipe stress limit is based on high suppression pool level of 21'3".

C. Correct- The Primary Containment Pressure Limit of 30 psig requires venting containment to avoid exceeding this limit. Above this value, containment vent dampers (HVR-AOD127, HVR-AOD128 may be unable to open against this high pressure condition. The inability to lower containment pressure can lead to containment failure and an uncontrolled release of radioactive material.

D. The pressure suppression pressure is a function of suppression pool level and is used to determine when emergency depressurization is required, not emergency venting of containment.

Technical Reference(s): R-STM-0057 Rev 4 Pg 16 of 69

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-HLO-517 Obj 2.5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.9

Comments:

Validation team recommended to change distractor B from "exceeding the containment design pressure limit" to current reading based on 30 psig being inclusive of 15 psig.

Revised distractor A to read "Standby Gas Filter Train damage due to exceeding design inlet pressure" and distractor B to read "SRV tailpipe damage due to exceeding the code allowable stresses"

QUESTION 12 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295025 EA2.02		IR 4.2

Ability to determine and/or interpret the following as they apply to High Reactor Pressure:

EA2.02 Reactor power

Proposed Question:

The following conditions exist during an ATWS:

• Reactor power	45%
• Reactor pressure	1025 psig
• Main Steam Isolation Valves and Drains	CLOSED
• Pressure band	950-1090 psig

The unit operator has been directed to control reactor pressure with Safety Relief Valves (SRV). How many SRVs are required to maintain reactor pressure in the prescribed band?

A. 1-2

B. 3-4

C. 5-6

D. 7-8

Proposed Answer: D.

Explanation:
A. Each SRV can pass a steam flow equivalent of 6-7% power. 1-2 SRVs can handle power levels between 6% and 14%.
B. Each SRV can pass a steam flow equivalent of 6-7% power. 3-4 SRVs can handle power levels between 18% and 285%.
C. Each SRV can pass a steam flow equivalent of 6-7% power. 5-6 SRVs can handle power levels between 30% and 42%.
D. Correct - Each SRV can pass a steam flow equivalent of 7-8% power. 6-7 SRVs can handle power levels between 42% and 56%.

Technical Reference(s): R-STM-0109 Rev 12, Pg 9 of 95

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0109 Obj 3b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑2

10 CFR Part 55 Content: 55.41.b.7

Comments:
Added “and Drains” to conditions
Removed overlap from distractors and answer

QUESTION 13 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	1	Group # 1		
	K/A #	295026 EA1.01			IR 4.1

Ability to operate and/or monitor the following as they apply to Suppression Pool High Water Temperature:

EA1.01 Suppression pool cooling

Proposed Question:

Following an ATWS, both loops of Residual Heat Removal are in the Suppression Pool Cooling mode with maximum flow through the heat exchangers.

- Reactor power <5%
- RPV Level -50 inches, slowly raising to normal band
- RPV Pressure 0 psig
- Suppression Pool Level 19 feet 11 inches
- Suppression Pool Temp 140°F
- RHR A in Sup Pool Cooling @ 5400 gpm, SWP flow @ 5500 gpm
- RHR B in Sup Pool Cooling @ 5600 gpm, SWP flow @ 5700 gpm

Both divisions of Standby Service Water are in service due to a loss of Normal Service Water.

Based on these conditions, which of the following should be of concern to the operator?

A. RHR B system flow has exceeded limits.

B. SWP flow has exceeded limits.

C. RHR pumps may experience air entrainment due to vortex limit concerns.

D. RHR pumps may experience cavitation due to NPSH concerns.

Proposed Answer: A.

Explanation:
A. Correct-RHR shell side flow limit is 5550 gpm.
B. SWP flow is less than 5800 gpm per loop limit.
C. Vortex limit concerns are at <10 feet SP Level
D. NPSH concerns begin at 160°F

Technical Reference(s): SOP-0031 Rev 320 Precaution 2.4.5; EOP-0001, Rev 26 Caution 5

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0511 Obj 6; RLP-STM-0204 Obj 8

Question Source: Bank; April 2010 Audit #13 Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b7

Comments:

QUESTION 14 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	1	Group # 1		
	K/A #	295027 G.2.4.4			IR 4.5

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question:

Given the following plant conditions:

- Reactor power 0% (all rods in)
- Reactor level +33 inches
- Reactor pressure 890 psig
- Drywell pressure 1.5 psid
- Drywell temperature 138°F
- Containment temperature 92°F
- Containment pressure 0.28 psig
- Corrected Annulus Differential Pressure -4.5 inches H_2O

Based on the above conditions, which of the following Emergency Operating Procedures require entry?

A. EOP-1 only

B. EOP-1 and EOP-2

C. EOP-2 only

D. EOP-2 and EOP-3

Proposed Answer: C.

Explanation:
A. EOP-1 not applicable. RPV Level >9.7 inches, RPV Press.<1094.7 psig, Drywell pressure<1.68 psid, All rods in.
B. EOP-1 not applicable. RPV Level >9.7 inches, RPV Press.<1094.7 psig, Drywell pressure<1.68 psid, All rods in.
C. Correct – Containment temperature is above the 90°F entry condition.
D. EOP-3 is not applicable. Annulus pressure<3.0 inches H_2O.

Technical Reference(s): EOP-1 Rev 26, EOP-2 Rev 15, EOP-3 Rev 16
Proposed references to be provided to applicants during examination: None
Learning Objective: R-LPOPS-HLO-512 Obj 4; RLP-HLO-514 Obj 4; RLP-HLO-515 Obj 3
Question Source: Bank # RBS-NRC-10
Question History: Last NRC Exam February 2003
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑2
10 CFR Part 55 Content: 55.41.b.10
Comments:
Revised stem conditions such that containment pressure is 0.28 psig and containment temperature is 92°F.

QUESTION 15 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295028 EK1.01		IR 3.5

Knowledge of the operational implications of the following concepts as they apply to High Drywell Temperature:

EK1.01 Reactor water level measurement

Proposed Question:

EOP-001A is being implemented due to ATWS conditions. Reactor power is above 5%. The following parameters are indicated in the Main Control Room:

• Reactor Pressure	450 psig
• Wide Range Level	-135 inches
• Upset Range Level	+6 inches
• Shutdown Range Level	+10 inches
• Narrow Range Level	0 inches
• Fuel Zone Range Level	-180 inches
• Drywell Temperature	310°F
• Containment Temperature	165°F

Which of the following Reactor Level instruments are valid for the current conditions?

A. Fuel Zone and Upset Range only

B. Upset Range and Wide Range only

C. Fuel Zone and Wide Range only

D. All Reactor Level instruments are invalid

Proposed Answer: C.

Explanation:
A. Fuel Zone use is allowed, but Upset Range is only valid down to 53 inches due to DW above 300°F.
B. Wide Range use is allowed, but Upset Range is only valid down to 53 inches due to DW above 300°F.
C. Correct – Fuel Zone is valid above -310 inches and Wide Range is valid above -142 inches.
D. Fuel Zone and Wide Range are both above the Minimum Indicated Water Level for the provided temperatures and Part 1 of Caution 1 is satisfied due to the provided RPV Pressure and Drywell Temperature being in the Safe Zone of the RPV Saturation Temperature Curve.

Technical Reference(s): EOP-0001, Rev 26 Caution 1
Proposed references to be provided to applicants during examination: EOP-0001 Caution 1
Learning Objective: R-LPOPS-HLO-511 Obj 6
Question Source: Bank – RBS-NRC-558
Question History: Last NRC Exam February 1999
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3
10 CFR Part 55 Content: 55.41.b.10
Comments:

QUESTION 16 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295031 EK3.01		IR 3.9

Knowledge of the reasons for the following responses as they apply to Reactor Low Water Level:

EK3.01 Automatic depressurization system actuation

Proposed Question:

The reason for actuation of the Automatic Depressurization System is to__________.

A. lower reactor pressure to allow recovery of reactor water level by low pressure injection systems

B. lower the driving head in the reactor pressure vessel to reduce inventory loss during a Loss of Coolant Accident.

C. limit the release of fission products due to isolation valve leak-by after a Loss of Coolant Accident

D. preclude exceeding the ASME Code limit for peak pressure in the reactor coolant pressure boundary

Proposed Answer: A.

Explanation:

A. Correct – The ADS system is designed to lower reactor pressure during a small break LOCA coincident with failure of High Pressure Core Spray to allow low pressure ECCS to restore reactor water level.

B. Although lowering reactor pressure does reduce inventory loss during a LOCA, this is not the reason for ADS actuation.

C. Although lowering reactor pressure does limit the release of fission products due to reduced driving head, this is not the reason for ADS actuation.

D. The safety relief valves (which are used as part of ADS) provide the stated function in this distracter, but this is not the reason for ADS actuation.

Technical Reference(s): RLP-STM-0202 Rev 2 Pg 4 of 36

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0202 Obj 1

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 17 Rev 1

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 1 Group # 1
K/A # 295037 EA2.03 IR 4.3

Ability to determine and/or interpret the following as they apply to Scram Condition Present and Reactor Power above APRM Downscale or Unknown:
EA2.03 SBLC tank level

Proposed Question:

During ATWS conditions, the CRS has directed injection with Standby Liquid Control (SLC). If the initial level of the SLC tank is 3350 gallons, what is the highest SLC tank level at which you would report to the CRS that Hot Shutdown Boron Weight has been injected?

A. 1769 gallons

B. 1819 gallons

C. 2674 gallons

D. 2724 gallons

Proposed Answer: C.

Explanation:
A. This is the correct value for Cold Shutdown Boron Weight; stem asked for Hot Shutdown Boron Weight.
B. This is an interpolated value for Cold Shutdown Boron Weight. Additionally, the stem asked for Hot Shutdown Boron Weight.
C. Correct - Using OSP-0053 Hard Card, initial tank level of 3300 gallons should be used based on note above table. Hot Shutdown Boron Weight is 69 lbs, so first column is used yielding 2674 gallons.
D. This is an interpolated value for Hot Shutdown Boron Weight. Interpolation is not allowed. The lower tank value must be used as annotated in the note above the table.

Technical Reference(s): R-STM-0201 Rev 7 Pg 7 of 37; OSP-0053 Rev 20 Pg 46 of 71

Proposed references to be provided to applicants during examination: OSP-0053 Rev 20 Pg 46 of 71

Learning Objective: RLP-STM-0201 Obj 1a

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑2

10 CFR Part 55 Content: 55.41.b.10

Comments:
Reworded the stem of the question to read " what is the highest SLC tank level at which you would report to the CRS that Hot Shutdown Boron Weight has been injected"

QUESTION 18 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295038 EK2.04		IR 3.9

Knowledge of the interrelations between High Off-Site Release Rate and the following:

EK2.04 Stack-gas monitoring system: Plant-specific

Proposed Question:

Which of the following radiation monitors is utilized in determining when a High Offsite Radioactive Release is occurring that requires entry into the Emergency Operating Procedures?

A. RMS-RE16, CONTAINMENT POST ACCIDENT MONITOR

B. RMS-RE118, TURBINE BLDG VENT

C. RMS-RE124, COND DEMIN OFFGAS VENTILATION

D. RMS-RE125, MAIN PLANT EXHAUST

Proposed Answer: D.

Explanation:
A. RMS-RE125, 5A(B) & 6A(B) are used in determining entry into EOP-0003 due to High Offsite Release.
B. RMS-RE125, 5A(B) & 6A(B) are used in determining entry into EOP-0003 due to High Offsite Release.
C. RMS-RE125, 5A(B) & 6A(B) are used in determining entry into EOP-0003 due to High Offsite Release.
D. Correct – RMS-RE125 in Alarm condition indicates a High Offsite Release and requires entry into EOP-0003.

Technical Reference(s): EOP-0003 Rev 16 ; EIP-2-001 Rev 24 Pg 21 of 158

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-515 Obj 2

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.11

Comments: Validation team suggested changing distractor A from RE110, Aux Bldg to one that would not enter EOPs for any reason; RMS-RE110 does have an EOP entry level.

QUESTION 19 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	600000 AA1.06		IR 3.0

Ability to operate and/or monitor the following as they apply to Plant Fire On Site:
AA1.06 Fire Alarm

Proposed Question:

Firewatch personnel have identified a fire and operated a fire pull station, FPS-4CB in the HCU East Area of the Reactor Building 114' el. Manipulation of this pull station will alarm ___(1)___ and ___(2)___ result in actuation of a suppression system.

A. (1) locally only; (2) will

B. (1) locally only; (2) will not

C. (1) locally and in the Main Control Room, (2) will not

D. (1) locally and in the Main Control Room, (2) will

Proposed Answer: C.

Explanation:
A. Annunciation will also occur in the MCR. There is no suppression associated with this pull station.
B. Annunciation will also occur in the MCR. Part 2 is correct.
C. Correct – Alarms will sound locally as well as in the MCR. There is no suppression associated with this pull station.
D. Part 1 is correct. There is no suppression associated with this pull station.

Technical Reference(s): R-STM-0250, Rev 6 Pg 47 of 65 ; Pre-Fire Strategy RB-114-004, Rev 3 Pg 2 of 9

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0250 Obj 3

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑4

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 20 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 700000 AK3.01		IR 3.9

Knowledge of the reasons for the following responses as they apply to Generator Voltage and Electric Grid Disturbances:
AK3.01 Reactor and turbine trip criteria

Proposed Question:

The reactor has scrammed and the Main Turbine has tripped from 100% power operation. The transient was initiated by a grid disturbance which caused a trip of the Main Generator.

Which of the following describes the reasons these trips occurred?

The turbine tripped to ___(1)___.
The reactor scrammed ___(2)___.

A. (1) provide protection from a turbine overspeed condition due to the sudden unloading of the generator.
(2) in anticipation of pressure and flux transients due to the tripping of the turbine.

B. (1) protect the Main Generator from excessive excitation due to the sudden unloading of the generator.
(2) in anticipation of pressure and flux transients due to the tripping of the turbine.

C. (1) provide protection from a turbine overspeed condition due to the sudden unloading of the generator.
(2) due to an RPV low water level condition resulting from the turbine trip.

D. (1) protect the Main Turbine from excessive vibration due to the sudden unloading of the generator.
(2) due to an RPV low water level condition resulting from the turbine trip.

Proposed Answer: A.

Explanation:
A. Correct - On any generator trip the turbine trips to avoid a turbine overspeed condition due to no load on the generator. The reactor scrams in anticipation of pressure and flux transients due to the tripping of the turbine.
B. The exciter field breaker opening protects the main generator from excessive excitation.
C. The reactor scram is in anticipation of pressure and flux transients due to the tripping of the turbine. Although the pressure rise from a turbine trip would cause a reduction in RPV level it would not initiate a reactor scram.
D. Turbine vibration is not affected following a generator trip and the turbine vibration trips are bypassed following startup requiring manual tripping on vibration. The reactor scram is in anticipation of pressure and flux transients due to the tripping of the turbine. Although the pressure rise from a turbine trip would cause a reduction in RPV level it would not initiate a reactor scram.

Technical Reference(s): R-STM-0110 Rev 10 Pg 7 of 50; R-STM-0508 Rev 6 Pg 13 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0509 Obj 1, 3; RLP-STM-0508 Obj 2

Question Source: Bank Question History: Last NRC Exam November 2012

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41b.5

Comments: Appeared on one of last 2 NRC exams (2 of 3)
Revised distractor B(1) and D(1) for plausibility

QUESTION 21 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	1	Group # 2		
	K/A #	295002 AK3.09			IR 3.2

Knowledge of the reasons for the following responses as they apply to Loss of Main Condenser Vacuum:
AK3.09 Reactor power reduction

Proposed Question:

During periods of deteriorating condenser vacuum, reactor power is reduced in accordance with AOP-0005, LOSS OF MAIN CONDENSER VACUUM/TRIP OF CIRCULATING WATER PUMP, in order to avoid ___________________________.

A. explosive hydrogen concentrations in the main condenser

B. a cooldown rate of the turbine rotor in excess of 125°F per hour

C. high airborne levels due to loss of SJAE loop seals

D. turbine blade damage due to buffeting

Proposed Answer: D.

Explanation:

A. Operation of the SJAE and Offgas system remove non-condensable from the condenser preventing a buildup of hydrogen.
B. A higher condenser pressure would cause heating of the turbine, not accelerate the cooldown.
C. Loop seals are of sufficient height to not be affected by the change in pressure.
D. Correct – As Turbine exhaust backpressure rise, excess windage will be experience resulting on buffeting and heat of the last stage turbine blading.

Technical Reference(s): R-STM-0110 Rev 10, Pg 15 of 50

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP005 Obj 6

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.5

Comments:

QUESTION 22 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 2	
	K/A #	295007 G.2.2.44		IR 4.2

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions for High Reactor Pressure.

Proposed Question:

The plant was operating at 100% power when a Main Turbine trip occurred. On H13-P601, the SRV LOW-LOW SET (LLS) LOGIC indication lights (both A & B) are all illuminated.

Based on the above conditions, the next opening of Safety Relief Valves due to high reactor pressure will occur at ___(1)___ psig. If the LLS LOGIC RESET pushbutton is depressed, the next opening of Safety Relief Valves will occur at ___(2)___ psig.

A. (1) 1133; (2) 1133

B. (1) 1133; (2) 1063

C. (1) 1063; (2) 1143

D. (1) 1063; (2) 1133

Proposed Answer: D.

Explanation:

A. 1133 is incorrect because the status lights indicate that LLS is active making the initial SRV opening pressure 1063 psig. Part 2 is correct.

B. 1133 is incorrect because the status lights indicate that LLS is active making the initial SRV opening pressure 1063 psig. Part 2 in incorrect because depressing the pushbutton restores the initial SRV opening setpoint to 1133 psig.

C. Part 2 is correct. 1143 psig is the opening pressure for subsequent SRV openings, not the initial opening following reset of the LLS logic.

D. Correct – All LLS illuminated indicates that LLS has actuated therefore the initial opening of SRVs will occur at 1063 psig. After the RESET pushbutton is depressed, the initial setpoint of 1133 psig will be restored.

Technical Reference(s): R-STM-0109 Rev 12 Pg 33 of 95 and Table 3

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0109 Obj 4, 10

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7

Comments: Validation team suggested changing the verb tense in the stem to current version (initial opening to next opening).

QUESTION 23 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	1	Group # 2	
	K/A #	295008 AK1.01		IR 4.0

Knowledge of the operational implications of the following concepts as they apply to High Containment Temperature (Mark III Containment Only):
AK1.01 Containment pressure: Mark III

Proposed Question:

Following a Loss of Coolant Accident (LOCA), drywell bypass leakage has resulted in the following conditions:

- Reactor Level -80 inches, stable
- HVR-UC1A & 1B running
- Containment Temperature 105°F, slowly rising
- Containment Pressure 0.85 psig, slowly rising

Which of the following actions can be performed to address the rising containment pressure?

A. Normal containment venting with HVR-FN14

B. Normal containment venting with Standby Gas Treatment

C. Restoration of chilled water (HVN) to the containment unit coolers

D. Alignment of Service Water (SWP) to the containment unit coolers

Proposed Answer: D.

Explanation:

A. Although venting with HVR-FN14 will lower containment pressure, it can not be performed below Level 2.

B. Although venting with Standby Gas Treatment will lower containment pressure, it can not be performed below Level 2.

C. HVN can not be restored due to Level 2 isolation signal present.

D. Correct - The only method of lowering containment pressure with the given conditions is to supply SWP as cooling water to the containment unit coolers.

Technical Reference(s): AOP-0003 Rev 33 Pg 15,18 of 22; SOP-0059 Rev 34 Pg 44 of 83

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0403 Obj 4.2

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.5

Comments: NRC questioned if there are any EOP attachments that can be implemented to make distractors A,B, or C correct? The answer is NO, there is no enclosure for overriding the level two isolation of the containment HVN valves and the only enclosure that allows normal venting is EOP-0005 Enclosure 25, DEFEATING PRIMARY CONTAINMENT VENT AND PURGE ISOLATION INTERLOCKS, which is only directed for use during performance of SAP-0002, SEVERE ACCIDENT PROCEDURE – CONTAINMENT AND RADIOACTIVE RELEASE CONTROL. Based on the conditions given, SAP-0002 would not be entered and therefore EOP-0005 Enclosure 25 would not be implemented.

QUESTION 24 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	1	Group #	2	
	K/A #	295015 AA2.02			IR 4.1

Ability to determine and/or interpret the following as they apply to Incomplete Scram:
AA2.02 Control rod position

Proposed Question:

A reactor scram, with a failure of all control rods to fully insert has occurred coincident with a failure of the Full Core Display.

In accordance with SOP-0071, ROD CONTROL AND INFORMATION SYSTEM, individual control rod position information is determined at ____________.

A. either Rod Action Control System (RACS) cabinet

B. RACS Cabinet 1 only

C. RACS Cabinet 2 only

D. Rod Gang Drive System (RGDS) cabinet

Proposed Answer: A.

Explanation:
A. Correct – Rod position information can be determined from either H13-P651 or H13-P652.
B. Information concerning rod position can be obtained from both RACS cabinets.
C. Information concerning rod position can be obtained from both RACS cabinets.
D. Information concerning rod position can be obtained from both RACS cabinets, but not from RGDS.

Technical Reference(s): SOP-0071 Rev 29 Pg 13 of 49; R-STM-0500 Rev 4 Pg 31 of 46

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0500 20b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.6

Comments:

QUESTION 25 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐		
	Tier #	1	Group #	2		
	K/A #	295029 EK2.06				IR 3.4

Knowledge of the interrelations between High Suppression Pool Water Level and the following:
EK 2.06 SRVs and discharge piping

Proposed Question:

The highest suppression pool level where operation of a Safety Relief Valve will not result in exceeding the code allowable stresses in the SRV tail pipe is _________.

A. 21’ 6”

B. 21’ 3”

C. 21’ 0”

D. 20’ 0”

Proposed Answer: B.

Explanation:
A. 21’6” is the Maximum Pressure Suppression Primary Containment Water Level
B. Correct – 21’3” defines the SRV Tail Pipe Level Limit as described in EPSTG0002.
C. 21’0” is the level which requires a reactor scram per EOP-0002.
D. 20’0” is the upper level of the Suppression Pool normal band.

Technical Reference(s): R-STM-0057 Rev 4 Pg 12, 16 of 69, EPSTG*0002 App A, A-55

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-517 Obj 3.21

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.9

Comments:

QUESTION 26 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 2	
	K/A #	295036 EA1.03		IR 2.8

Ability to operate and/or monitor the following as they apply to Secondary Containment High Sump/Area Water Level:
EA1.03 Radwaste

Proposed Question:

Following a plant scram, EOP-0003, SECONDARY CONTAINMENT AND RADIOACTIVE RELEASE CONTROL, has been entered due to high level in DFR-TK5A and TK5B, AUXILIARY BLDG FLOOR DRAIN SUMPS.
No operator action has been taken to address the sump level issue.

Based on the above conditions, level in the ___________ will rise as pumps associated with the 2 tanks automatically run to lower level in the sumps.

A. Suppression Pool

B. Waste Collector Tank

C. Floor Drain Collector Tank

D. Regenerant Waste Tank

Proposed Answer: C.

Explanation:
A. Contents of these tank can be pumped back to the suppression pool, but manual operator action is required to open DFR-MOV146. The stem states that no operator action has been taken.
B. WC Tank receives equipment drain waste. Tank 5A and 5B pump to Floor Drains in Radwaste.
C. Correct – With suppression pool pumpback closed (no operator action taken), the content of these tanks is pumped to the Floor Drain Collect tank in Radwaste.
D. RW tanks receive water from Chem labs, sample panels and URC systems.

Technical Reference(s): R-STM-0609 Rev 5 Pg 14 of 69.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0609 Obj 2b, 3e, 4b, 7a

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑4

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 27 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	1	Group #	2	
	K/A #	500000 EK3.03			IR 3.0

Knowledge of the reasons for the following responses as they apply to High Primary Containment Hydrogen Concentrations:
EK3.03 Operation of hydrogen and oxygen recombiners

Proposed Question:

The Hydrogen Recombiners are secured when containment hydrogen concentrations exceed the Hydrogen Deflagration Overpressure Limit (HDOL) because _______________.

A. insufficient amounts of are oxygen available to support recombination

B. the heat produced from recombination at this concentration will overload cooling capability of containment unit coolers

C. the hydrogen igniters alone are a sufficient method of hydrogen removal

D. the recombiner may become an ignition source leading to rapid uncontrolled combustion

Proposed Answer: D.

Explanation:
A. See “D”.
B. See “D”.
C. See “D”.
D. Correct - When the HDOL is exceeded, this indicates that containment can be over pressurized if a deflagration event were to occur. To avoid deflagration all potential ignition sources are removed (igniters, recombiners) and containment purge is the only authorized method of hydrogen removal.

Technical Reference(s): EPSTG*0002 App A, A-15 ; App B, B-14-13 & B-14-23

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-517 Obj 2.4

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 28 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	203000 K5.02		IR 3.5

Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: Injection Mode (Plant Specific):
K5.02 Core cooling methods

Proposed Question:

A plant transient has resulted in the following conditions:

RPV Level -180 inches
RHR A is the only available injection source and is injecting 5000 gpm.

Based on the given conditions, the adequate core cooling method in effect is _________

A. submergence.

B. steam cooling with injection.

C. steam cooling without injection.

D. spray cooling.

Proposed Answer: B.

Explanation:
A. Adequate core cooling by submergence required level >-162”,
B. Correct. With injection present, the fuel can be cooled by steam at level above 186”.
C. At -180”, level is above the minimum zero injection reactor water level limit (-200”), but this level is only valid with no injection into the core. Per the stem, RHR A is injection 5000 gpm.
D. Only HPCS/LPCS can provide spray cooling if they are injecting at least 5000 gpm. RHR A injects through a nozzle not a spray sparger.

Technical Reference(s): EOP-0001 Rev 26 ACC Chart

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-511 Obj.3, 4

Question Source: Bank; Nov 2010 Audit #28

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑2

10 CFR Part 55 Content: 55.41b.5

Comments:

QUESTION 29 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	2	Group #	1	
	K/A #	205000 K2.01			IR 3.1

Knowledge of the electrical power supplies to the following:
K2.01 Pump motors

Proposed Question:

The power supply to Residual Heat Removal (RHR) Pump “B” is _____________.

A. ENS-SWG1A

B. ENS-SWG1B

C. NNS-SWG1A

D. NNS-SWG1B

Proposed Answer: B.

Explanation:
A. ENS-SWG1A is the power supply to RHR Pump “A”
B. Correct – ENS-SWG1B is the power supply to RHR “B”
C. Although a 4160VAC bus, this bus is non-safety related.
D. Although a 4160VAC bus, this bus is non-safety related.

Technical Reference(s): SOP-0031, Rev 320 Pg 141 of 187

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204 Obj. 11c

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 30 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	2	Group # 1	
	K/A #	209001 A2.06		IR 3.2

Ability to (a) predict the impacts of the following on the Low Pressure Core Spray System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
A2.06 Inadequate system flow

Proposed Question:

A small break LOCA has resulted in a High Drywell Pressure initiation signal. All ECCS pumps have started. RHR A has been aligned in the Suppression Pool Cooling mode but no other operator action has been taken. The unit operator observes the following:

- Reactor Pressure — 750 psig
- Suppression Pool Level — 15 feet 3 inches
- Suppression Pool Temp — 101°F
- E21-MOVF011, LPCS MIN FLOW VLV — CLOSED
- E21-MOVF012, LPCS TEST RETURN VLV — CLOSED
- E12-MOVF024A, RHR PUMP A TEST RET TO SUP PL — OPEN

(1) What is the impact of continued operation in this condition and (2) What action should the operator take to mitigate this condition?

A. (1) LPCS Pump degradation due to inadequate flow; (2) Open the minimum flow valve

B. (1) LPCS Pump degradation due to inadequate flow; (2) Open the Test Return to the Suppression Pool.

C. (1) ECCS pump cavitation due to inadequate net positive suction head; (2) Raise suppression pool level.

D. (1) ECCS pump cavitation due to inadequate net positive suction head; (2) Maximize suppression pool cooling with all available systems.

Proposed Answer: A.

Explanation:

A. Correct – The LPCS is dead-headed. Per P&L 2.2 after 60 seconds of operation in this condition pump degradation will be experienced. The minimum flow must be opened to establish flow.

B. Part 1 is correct, but the Test Return to the Suppr Pool can't be opened due to the RHR A alignment. Ref. P&L 2.15

C. Although suppression pool temperature is elevated and suppression pool level is below normal, cavitation is not expected to occur at temperatures below 160°F and suppression pool levels above 10 feet,

D. See "C",

Technical Reference(s): SOP-0032 Rev 22 Page 3 and 4 of 29

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205 Obj 5, 8

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.5 Comments:

QUESTION 31 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	209001 A4.07		IR 2.7

Ability to manually operate and/or monitor in the control room:
A4.07 Fill pump (LPCS)

Proposed Question:

The unit operator observed the following indications on H13-P601 for
E21-C002 LPCS/RHR DIV 1 LINE FILL PUMP:

Green Light	ILLUMINATED
Amber Light	ILLUMINATED
Red Light	OFF

Following receipt of these indications, which of the following alarms should the operator anticipate?

A. LPCS/LPCI A INJ VALVE RPV PRESS LOW AND
LPCS INJECTION LINE PRESSURE HI/LOW

B. RHR PUMP A DISCH PRESSURE HI/LOW AND
LPCS PUMP E21-C001 IN MAUAL OVERRIDE

C. LPCS INJECTION LINE PRESSURE HI/LOW AND
RHR PUMP A DISCH PRESSURE HI/LOW

D. LPCS PUMP E21-C001 IN MAUAL OVERRIDE AND
LPCS/LPCI A INJ VALVE RPV PRESS LOW

Proposed Answer: C.

Explanation:
A. The second alarm is correct, but the first alarm would not be present as a result of a tripped line fill pump.
B. First alarm is correct, but second would not be present as a result of a tripped line fill pump, the amber light would be lit if the main pump was in manual override which adds to the plausibility of this distractor.
C. Correct - The indications provide are that of a tripped line fill pump. This condition will result in a low pressure condition in the injection lines of LPCS and RHR A which have a common line fill system.
D. Neither of these alarms would be present as a result of a tripped line fill pump., the amber light would be lit if the main pump was in manual override which adds to the plausibility of this distractor.

Technical Reference(s): ARP-601-20 Rev 305 Pg 19 of 38 & ARP-601-21 Rev 312 Pg 31 of 79

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205 Obj 6, 7, 11f, 19

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 32 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	209002 K1.04		IR 3.8

Knowledge of the physical connections and/or cause-effect relationships between High Pressure Core Spray System (HPCS) and the following:
K1.04 HPCS diesel generator: BWR-5,6

Proposed Question:

The High Pressure Core Spray (HPCS) Diesel Generator is running and paralleled to offsite power in support of STP-309-0613, DIVISION III DIESEL GENERATOR 24 HOUR RUN.

A loss of offsite power occurs resulting in a loss of power to the electrical distribution system switchgear NNS-SWG1C with a concurrent Loss of Coolant Accident (LOCA).

Which of the following represents the resulting status of the individual High Pressure Core Spray system components and closure time of the HPCS Pump Breaker?

	E22-ACB01, HPCS D/G OUTPUT BRKR	E22-ACB02, HPCS PUMP BRKR (breaker closure time)	E22-MOVF012, HPCS MIN FLOW VALVE TO SUPPRESSION POOL
A.	Closed	immediately	Closed
B.	Closed	30 seconds	Open
C.	Open	immediately	Closed
D.	Open	30 seconds	Open

Proposed Answer: A.

Explanation:

A. Correct – Output was initially closed and will remain closed. The pump breaker will close immediately and the minimum flow valve will be closed because flow will be maintained through the injection valve.

B. Load shedding does not occur because bus voltage was never lost. Additionally, 30 second sequencer applies to SWP-P2C which supplies cooling to the Div 3 DG. HPCS pump breaker closure is immediate.

C. DG output breaker remains closed to supply the bus due to the loss of offsite power.

D. See C; Additionally, 30 second sequencer applies to SWP-P2C which supplies cooling to the Div 3 DG. HPCS pump breaker closure is immediate.

Technical Reference(s): R-STM-0309H Rev 12, Pg 4, 48, 49 of 78

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0203 Obj 13d, RLP-STM-0309H Obj 15f

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑4

10 CFR Part 55 Content: 55.41.b.8

Comments: Revised STEM to read “resulting in a loss of power to the electrical distribution system switchgear NNS-SWG1C” to address STM scenario

QUESTION 33 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	2	Group # 1	
	K/A #	211000 A1.09		IR 4.0

Ability to predict and/or monitor changes in parameters associated with operating the Standby Liquid Control System controls including:
A1.09 SBLC system lineup

Proposed Question:

During an ATWS condition, Standby Liquid Control "A" was initiated with SLC tank level initially at 3800 gallons. 10 minutes later the following conditions exist:

SLC Tank Level	3800 gallons
SLC discharge pressure	1400 psig

SLC Pump A was secured and SLC Pump B was initiated. 10 minutes later the following conditions exist:

SLC Tank Level	3800 gallons
SLC discharge pressure	1400 psig

Which of the following is responsible for both sets of indications above? (Assume a single failure)

A. A Squib valve failure

B. Suction valve failure

C. Injection line manual isolation valve left closed following maintenance

D. Test tank inlet valve left closed following maintenance

Proposed Answer: C.

Explanation:
A. A single squib valve failure would not affect both subsystems.
B. A single suction valve failure would not affect both subsystems.
C. Correct – The closure of the common injection line manual isolation would produce the indicated parameters due to "dead-heading" the system.
D. The test tank inlet valve is normally closed and therefore would not produce the given indications.

Technical Reference(s): PID 27-16A

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0201 Obj 2, 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.5

Comments: Revised distractor A to read "A squib valve failure" to distinguish between one or both squib valves failing.

QUESTION 34 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	212000 K3.06		IR 4.0

Knowledge of the effect that a loss or malfunction of the Reactor Protection System will have on the following:
K3.06 Scram air header solenoid operated valves

Proposed Question:

The plant is operating at 100% power when a loss of Reactor Protection System (RPS) "A" occurs. RPS "B" is unaffected and remains energized.

This results in the 145 scram pilot valves ___(1)___ and the backup scram valves ___(2)___.

A. (1) repositioning;
(2) de-energizing

B. (1) remaining unchanged;
(2) remaining de-energized

C. (1) repositioning;
(2) energizing

D. (1) remaining unchanged;
(2) remaining energized

Proposed Answer: B.

Explanation:

A. The scram pilot valve will not reposition on a single solenoid being de-energized and the backup scram valves are normal de-energized.

B. Correct – The "A" solenoid on the scram pilot valves will de-energize but the pilot valve will not reposition as long as the "B" solenoid is still energize. The backup scram valve are normally de-energize and will remain as such with the loss of only RPS A.

C. The scram pilot valve will not reposition on a single solenoid being de-energized and the backup scram valves require loss of both RPS A and B to energize.

D. Part 1 is correct, but the backup scram valves are normally de-energize and will remain de-energize with the loss of only one RPS bus.

Technical Reference(s): R-STM-0508 Rev 6 Pg 20-22 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508 Obj 8d

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.6

Comments:

QUESTION 35 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	212000 G.2.1.28		IR 4.1

Knowledge of the purpose and function of major Reactor Protection System components and controls.

Proposed Question:

With the Reactor Mode Switch in the RUN position, which of the following represents the status of the indicated interlocks?

	IRM Hi Flux Scram	MSIV Closure Scram	RPV Hi Water Level Scram
A.	Enabled	Bypassed	Bypassed
B.	Bypassed	Enabled	Bypassed
C.	Enabled	Bypassed	Enabled
D.	Bypassed	Enabled	Enabled

Proposed Answer: D.

Explanation:

A. IRM Scram is not enabled in RUN, nor is the MSIV Closure is not bypassed, nor is the high level scram bypassed.

B. IRM scram is bypassed and the MSIV closure scram is enabled, but the high level scram is not bypassed.

C. IRM scram is not enabled in RUN, nor is the MSIV scram bypassed, but the high level scram is enabled.

D. Corrrect – The IRM scram is bypassed in RUN and the MSIV closure scram is enabled, and the high water level scram is enabled.

Technical Reference(s): R-STM-0508 Rev 6 Pg 57 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508 Obj 3b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.6

Comments:

QUESTION 36 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	2	Group # 1		
	K/A #	215003 K5.01			IR 2.6

Knowledge of the operational implications of the following concepts as they apply to Intermediate Range Monitor (IRM) System:
K5.01 Detector operation

Proposed Question:

During a plant startup the high voltage power supply for IRM 'D" fails and drops to 0 volts. The Reactor Mode Switch is in the START / HOT STANDBY position and all other IRMs are OPERABLE.

How will the RC&IS and RPS respond to the change in IRM Channel "D" detector supply voltage?

A. BOTH a control rod withdrawal block and a RPS half scram signal are generated.

B. ONLY a control rod withdrawal block signal is generated.

C. ONLY a RPS half scram signal is generated

D. NEITHER a control rod withdrawal block NOR a RPS half scram signal is generated

Proposed Answer: A.

Explanation:
A. Correct – Both the IRM Rod Block and RPS trip signals are generated due to low voltage to the detector.
B. RPS trip signal will also be generated.
C. RC&IS Rod Block will also be generated.
D. With the Mode Switch in RUN this would be correct, but with the Mode Switch in Start/Hot Stby both RPS trip and RC&IS rod block will be generated.

Technical Reference(s): R-STM-0503 Rev 7 Pg 100-101 of 111

Proposed references to be provided to applicants during examination: None

Learning Objective: R-STM-0503 Rev 7 Obj 13

Question Source: Bank # RBS-NRC-1

Question History: Last NRC Exam February 2003

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.2

Comments:

QUESTION 37 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	2	Group # 1		
	K/A #	215004 K6.01			IR 3.2

Knowledge of the effect that a loss or malfunction of the following will have on the Source Range Monitor (SRM) System:
K6.01 RPS

Proposed Question:

A plant startup is in progress and the reactor is nearing criticality when RPS Motor Generator Set A generator output breaker trips.

What is the status of the SRM A indicating lights on H13-P680?

A. The UPSC TRIP, ALARM OR INOP, and DNSC lights will ALL be lit.

B. ONLY the ALARM OR INOP, and DNSC lights will be lit.

C. ONLY the ALARM OR INOP light will be lit.

D. NONE of the indicating lights will be lit.

Proposed Answer: A.

Explanation:
A. Correct - All indicating lights illuminate on loss of power.
B. All indicating lights illuminate on loss of power.
C. All indicating lights illuminate on loss of power.
D. All indicating lights illuminate on loss of power.

Technical Reference(s): AOP-0010 Rev 19, Step 2.1 Pg 3 of 21

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503 Obj 7, 35

Question Source: Bank # RBS-NRC-908

Question History: Last NRC Exam Sept 2004 #35

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 38 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	215005 K1.09		IR3.6

Knowledge of the physical connections and/or cause-effect relationships between Average Power Range Monitor/Local Power Range Monitor System and the following:
K1.09 Reactor recirculation system: BWR-5,6

Proposed Question:

The Average Power Range Monitors (APRM) Upscale Thermal trip is biased by a Reactor Recirculation flow signal generated from differential pressure signals measured across ______________.

A. the core plate

B. the recirculation loop elbows

C. calibrated jet pumps

D. all jet pumps

Proposed Answer: B.

Explanation:
A. Although core flow can be calculated from core plate differential pressure, this signal is not input to the APRMs.
B. Correct – The Recirculation Flow Monitoring Subsystem consists of flow converters which receive a differential pressure signal provided from elbow taps on both recirculation loops.
C. Although core flow can be calculated from calibrated jet pump flows, this signal is not input to the APRMs.
D. Although core flow can be calculated from total jet pump flow, this signal is not input to the APRMs..

Technical Reference(s): R-STM-0503 Rev 7 Pg 61 of 111

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503 Obj 27

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 39 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	217000 A3.03		IR 3.7

Ability to monitor automatic operations of the Reactor Core Isolation Cooling System (RCIC) including:
A3.03 System pressure

Proposed Question:

The plant is recovering from a loss of Offsite Power. Reactor Core Isolation Cooling (RCIC) is injecting in the reactor vessel as efforts are made to cool down the reactor pressure vessel to Cold Shutdown conditions.

What is the expected response of RCIC parameters as reactor pressure reduction begins?

	System flow	Discharge Pressure
A.	Lowers	rises
B.	Remains constant	rises
C.	Rises	lowers
D.	Remains constant	lowers

Proposed Answer: D.

Explanation:
A. Flow remains constant at 600 gpm, Discharge pressure lowers as reactor pressure lowers.
B. Flow does remain constant, but discharge pressure lowers as reactor pressure lowers.
C. Flow remains constant at 600 gpm, Discharge pressure does lower.
D. Correct- Flow remains constant at 600 gpm, discharge pressure lowers as reactor pressure lowers.

Technical Reference(s): R-STM-0209 Rev 10 Pg 10 of 52

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209 Obj 12

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑4

10 CFR Part 55 Content: 55.41.b.7

Comments: Eliminated turbine speed from the answer and distractors.

QUESTION 40 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	218000 K6.07		IR 3.4

Knowledge of the effect that a loss or malfunction of the following will have on the Automatic Depressurization System:
K6.07 Primary containment instrumentation.

Proposed Question:

A LOCA has occurred inside the drywell; however, the "A" high drywell pressure input into Automatic Depressurization System (ADS) logic channel "A" failed low. Conditions are as follows:

- Drywell pressure is 5 psid
- RHR "A", "B" and "C" are operating in the LPCI mode
- Reactor water level is at Level 2 and decreasing

Upon reaching Level 1, _______________________.

A. ADS valves will <u>NOT</u> open

B. ADS valves will open immediately

C. ADS valves will open after a 105 second time delay

D. ADS valves will open after a 5 minute + 105 second time delay

Proposed Answer: C.

Explanation:
A. A single instrument failure will not prevent the ADS system from actuating on a valid signal. The Div 2 solenoid will energize to open all ADS valves.
B. The logic will not be satisfied immediately. The 105 second timer must be timed out to complete the logic.
C. Correct-The failed instrument will not prevent the logic from being complete since the Div 2 solenoid will energize to open the ADS valve.
D. Even though the 5 minute timer does bypass the drywell pressure signal, the Div 2 logic is not affected by the Div 1 failure, so the expiration of the 5 minute timer is not required to open the valves.

Technical Reference(s): R-STM-0202 Rev 2 Pg 12 of 36

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0202 Obj 12.6

Question Source: Bank # 2007 NRC Exam # 41

Question History: Last NRC Exam RBS 2007 NRC Exam

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 41 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	2	Group #	1	
	K/A #	223002 K4.08			IR 3.3

Knowledge of Primary Containment Isolation System/Nuclear Steam Supply Shut-Off design feature(s) and/or interlocks which provide for the following:
K4.08 Manual defeating of selected isolations during specified emergency conditions.

Proposed Question:

During an ATWS, the following conditions exist:

- RPV Level -65 inches, steady
- Drywell pressure 1.52 psid, lowering

The CRS has directed the defeating of instrument air isolation interlocks. This task is accomplished by ________________________.

A. depressing the Inboard and Outboard isolation seal-in reset pushbuttons

B. placing the associated keylock switch to Emergency

C. removing the associated relay

D. lifting associated leads

Proposed Answer: B.

Explanation:
A. Due to level being <-43 inches, the logic can not be reset with the reset pushbuttons.
B. Correct – EOP Enclosure 16 provides guidance under these conditions to place the associated keylock switch to Emergency.
C. Although some Enclosures defeat isolation interlocks by removing relays, the IAS interlock is bypassed via a keylock switch.
D. See C.

Technical Reference(s): EOP-0005 Rev 314 Pg 59 of 139

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-516 Obj 1

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 42 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	239002 A4.02		IR 3.6

Ability to manually operate and/or monitor in the control room:
A4.02 Tail pipe temperatures

Proposed Question:

With the plant operating at 100% power, the ADS/SRV VALVE LEAKING annunciator alarms in the Main Control Room.

Which of the following ranges of temperatures include the tailpipe temperature that the unit operator would expect to see on B21-R614 Temperature Recorder for the open Safety Relief Valve?

A. 500°F - 575°F

B. 425°F - 499°F

C. 350°F - 424°F

D. 275°F - 349°F

Proposed Answer: D.

Explanation:
A. See "D".
B. See "D".
C. See "D".
D. Correct - At 100% power, reactor pressure is slightly over 1000 psia. Since expansion through throttle valve is assumed to be an isenthalpic event, using a Mollier diagram and from 1000 psia on the saturation curve moving on the line of constant enthalpy to the left towards atmospheric conditions yields at temperature of approximately 320°F.

Technical Reference(s): Mollier Diagram

Proposed references to be provided to applicants during examination: Steam Tables and Mollier Diagram

Learning Objective: RLP-STM-0109 Obj 26a

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑2

10 CFR Part 55 Content: 55.41.b.7

Comments: Revised distractor C to read "350°F - 424°F"

QUESTION 43 Rev 2

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	2	Group #	1	
	K/A #	259002 K4.10			IR 3.4

Knowledge of Reactor Water Level Control System design feature(s) and/or interlocks which provide for the following:
K4.10 Three element control (main steam flow, reactor feedwater flow, and reactor water level provide input)

Proposed Question:

The Feed Water Level Control System (FWLCS) is designed so that it can be operated using either three input signals or a single input signal to control reactor water level.

The FWLCS three element control feature is not used at low power because the ____.

A. dynamic compensator is too sensitive

B. feed flow output voltages are too low

C. steam flow instrumentation is not accurate

D. feed flow/steam flow summer reaction time is too sluggish

Proposed Answer: C.

Explanation:

A. The dynamic compensator receives the output from the feed flow/steam flow summer, and its purpose is to smooth the control signal such that the FWLC system is not too sensitive, minimizing “hunting”.

B. Instrument circuitry is calibrated to process signals over a given range.

C. Correct – With Reactor power below 15% Steam Flow and Feed Flow, are not accurate enough to predict level changes.

D. Steam flow instrument is not accurate at low power; see A.

Technical Reference(s): R-STM-0107 Rev 26 Pg 61 of 102

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107B Obj 14d

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7

Comments: Replaced Question;

QUESTION 44 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	2	Group # 1	
	K/A #	259002 G.2.1.32		IR 3.8

Ability to explain and apply system limits and precautions of Reactor Water Level Control System.

Proposed Question:

During power operations, the following Feedwater System (FWS) conditions exist:

- FWS-P1A — RUNNING, 355 amps
- FWS-P1B — OFF
- FWS-P1C — OFF
- Master Level Controller — AUTO
- FWS Regulating Valve A — AUTO, 90% open
- FWS Regulating Valve B — MANUAL, 20% open
- FWS Regulating Valve C — MANUAL, 0% open
- Startup Regulating Valve — AUTO, 100% open

Based on the above conditions, which of the following should be of concern?

A. Operating FWS pump amperage loading indicates excessive load on the pump.

B. Startup FWS Regulating Valve has exceeded flow limitations.

C. Operating FWS pump amperage loading indicates a minimum flow valve malfunction.

D. In service FWS Regulating Valve has inadequate margin for valve modulation.

Proposed Answer: A.

Explanation:
A. Correct - Per SOP-0009 P&L 2.14, full load current should be limited to 311 amps, however operation abvove 311 has been evaluated and pre-approved as long as amps are limited to less than 350 amps; FWS-P1A is above the limit.
B. Per SOP-0009, P&L 2.18 10% limit only applicable when startup reg valve is controlling RPV l;evel alone.
C. Per SOP-0009, P&L 2.9 minimum flow valve malfunction is indicated if <200 amps
D. Per SOP-0009, P&L 2.22, FWS Regulating Valve position should be limited to ≤92% open to allow an adequate margin for valve modulation while maintaining reactor level.

Technical Reference(s): SOP-0009 Rev 62, Pg 5-8 of 99

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107A Obj 9 & RLP-STM-0107B Obj 11

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.4

Comments:

QUESTION 45 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	2	Group # 1	
	K/A #	261000 A1.02		IR 3.1

Ability to predict and/or monitor changes in parameters associated with operating the Standby Gas Treatment System controls including:
A1.02 Primary containment pressure

Proposed Question:

Standby Gas Treatment Train 'A' is running in the Containment Purge lineup. Containment pressure is stable.

Which of the following describes the effect on containment pressure if the 'A' Standby Gas Treatment Train STOP pushbutton is depressed?

A. Containment Pressure will rise due to supply air from HVR-FN8, HIGH VOL CONTMT PURGE

B. Containment Pressure will rise due to supply air from HVR-FN13, LOW VOL CONTMT PURGE

C. Containment Pressure will remain stable as HVR-FN8, HIGH VOL CONTMT PURGE trips

D. Containment Pressure will remain stable as HVR-FN13, LOW VOL CONTMT PURGE trips

Proposed Answer: A.

Explanation:
A. Correct - When GTS runs in the containment purge lineup, its associated supply fan is HVR-FN8. Both these fans are 12,500 scfm capacity fans. If GTS A is secured, HVR-FN8 will continue to supply 12,500 scfm air to containment resulting in rising pressure.
B. HVR-FN13 is the low volume supply fan for containment purge and would not be run in conjuction with GTS.
C. HVR-FN8 will not trip when GTS is stopped.
D. HVR-FN13 would not be run in conjuction with GTS, nor would it trip if its associated exhaust fan (HVR-FN14) were to be secured when in the Containment Low Volume purge lineup.

Technical Reference(s): R-STM-0257 Rev 5, pg 13 of 28, R-STM-0403 Rev 8, pg 18-21 of 50.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0257 Obj 7 & 11c

Question Source: Bank; December 2010 NRC Exam # 46

Question History: Last NRC Exam December 2010

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41b.5

Comments: Appeared on one of last 2 exams (3 of 3).
Replaced bullets in stem with full component descriptions

QUESTION 46 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	2	Group # 1	
	K/A #	262001 K3.01		IR 3.5

Knowledge of the effect that a loss or malfunction of the A.C. Electrical Distribution will have on the following:
K3.01 Major system loads

Proposed Question:

A failure of NPS-SWG1B has occurred while operating at 90% power.

Which of the following lists the condensate and feedwater pumps that <u>remain running</u> after this event?

A. Condensate pump B; Feedwater pumps B and C

B. Condensate pumps A and C; Feedwater pump A

C. Condensate pump B; Feedwater pumps A and C

D. Condensate pumps A and B; Feedwater pump C

Proposed Answer: B.

Explanation:
A. Condensate B and Feedwater B and C would all be tripped on a loss of NPS-SWG1B.
B. Correct – Condensate A and C as well as Feedwater A are all powered from NPS-SWG1A and would remain running on a loss of NPS-SWGR1B.
C. Condensate B and Feedwater C would both trip on loss of NPS-SWGR1B. Feedwater A would remain running.
D Condensate B and Feedwater C would both trip on loss of NPS-SWGR1B. Condensate A would remain running.

Technical Reference(s): R-STM-0107 Rev 27 Pg 7 of 101 and R-STM-0104 Rev 7 Pg 13 of 65

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-107A Obj 4a and RLP-STM-0104 Obj 4

Question Source: Bank # RBS-NRC-362

Question History: Last NRC Exam January 1994

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41b.7

Comments:

QUESTION 47 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	262002 A3.01		IR 2.8

Ability to monitor automatic operations of the Uninterruptable Power Supply (A.C./D.C.) including:

A3.01 Transfer from preferred to alternate source.

Proposed Question:

ENB-INV01B has experienced a condition which caused power to be supplied from the alternate AC source through the static switch. How will this condition be detected?

A. An annunciator will alarm in the main control room and status light indication for the static switch at the inverter will illuminate.

B. An annunciator will alarm in the main control room and status light indication for the static switch on H13-P808 will illuminate.

C. This condition must be detected by the control building operator based on static switch indication at the inverter and reported to the control room due to no annunciator for this condition.

D. An alarm light will illuminate at the inverter and status light indication for the static switch on H13-P808 will illuminate.

Proposed Answer: A.

Explanation:

A. Correct - An ENB-INV01B alarm will annunciate in the MCR indicating inverter trouble. Local verification at the inverter will identify the specific trouble condition as static switch swap.

B. There is no method of verifying the static switch condition from H13-P808.

C. Although static switch position can be detected by the local operator, a control room annunciator will still alarm.

D. There is no method of verifying the static switch condition from H13-P808.

Technical Reference(s): R-STM-0300 Rev 26 Figure 7 ; ARP-808-87A, Rev 24 Pg 10 of 50

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0300 Obj 5, 12

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 48 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	263000 A1.01		IR 2.5

Ability to predict and/or monitor changes in parameters associated with operating the D.C. Electrical Distribution controls including:
A1.01 Battery charging/discharging rate

Proposed Question:

A 125 volt DC bus has experienced a trip of its battery charger supply breaker. Efforts to restore the battery charger to service have been unsuccessful. Assuming battery loading remains unchanged, predict the change in bus voltage under these conditions.

Voltage will _________________.

A. decrease in a linear fashion

B. slowly lower and then drop sharply

C. stair step down as individual cells are depleted

D. decrease sharply and then level off

Proposed Answer: B.

Explanation:
A. See “B”.
B. Correct – Battery voltage will decrease slowly then drop sharply as battery voltage reversal occurs. Reference SER 3-99.
C. See “B”.
D. See “B”.

Technical Reference(s): RPPT-STM-0305-ILO Rev 2, Slide 95 ; SER 3-99

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0305 Obj 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.8

Comments:

QUESTION 49 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	263000 G.2.4.31		IR 4.2

Knowledge of annunciator alarms, indications, or response procedures for DC Electrical Distribution.

Proposed Question:

The following control room annunciator is in alarm:

- 125VDC BATTERY CHGR ENB-CHR1B TROUBLE

On H13-P808, ENB-SWG01B:

- BUS VOLTAGE 126VDC
- BUS CURRENT 380 amps

At ENB-CHGR1B control panel:

- DC Voltmeter 126VDC
- DC Ammeter 360 amps
- Timer switch is at 0
- AC ON green light is lit

Assuming no operator action is taken, which of the following describes the expected ENB-SWG01B bus voltage trend and the reason for that trend?

The bus voltage will…

A. rise because an equalize charge is in progress.

B. lower because the bus load exceeds the charger's capacity.

C. lower because a malfunction of the float charge is indicated.

D. lower because AC power is not being supplied to the charger.

Proposed Answer: B.

Explanation:
A. With the timer at 0, an equalize charge is not in progress.
B. Correct – Load on the switchgear is 380 amps. Charger capacity is on 360 amps, so voltage will lower.
C. Elevated bus amps indicate a loading issue.
D. AC ON light indicates power available to the charger.

Technical Reference(s): R-STM-0305, Rev 6 page 7-8 of 37
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0305 Obj._4
Question Source: Bank # RBS-NRC-1107
Question History: Last NRC Exam RBS Feb 2003
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑4
10 CFR Part 55 Content: 55.41b.7 Comments:

QUESTION 50 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	2	Group # 1	
	K/A #	264000 K1.04		IR 3.2

Knowledge of the physical connections and/or cause-effect relationships between Emergency Generators (Diesel/Jet) and the following:
K1.04 Emergency generator cooling water system.

Proposed Question:

The plant has just experienced a loss of all offsite electrical power sources. The diesel generator status is as follows:

- Division 1 tagged out for super outage
- Division 2 tripped during start
- Division 3 running and supplying associated electrical bus

Under these conditions, cooling water is being supplied to the operating diesel generator by ___(1)___ and returns to the Standby Cooling Tower through ___(2)___.

A. (1) SWP-P2A; (2) SWP-MOV55A, STBY CLG TOWER 1 INLET

B. (1) SWP-P2A ; (2) SWP-AOV599, STBY CLG TWR INLET

C. (1) SWP-P2C; (2) SWP-MOV55A, STBY CLG TOWER 1 INLET

D. (1) SWP-P2C ; (2) SWP-AOV599, STBY CLG TWR INLET

Proposed Answer: D.

Explanation:

A. SWP-P2A would normally supply cooling water to the Division 1 loop but due to the loss of power and the diesel generator being tagged out, SWP-P2A cannot supply water to the Div 1 loop. Additionally, due to the loss of Div 1 power SWP-MOV55A cannot open.

B. SWP-P2A would normally supply cooling water to the Division 1 loop but due to the loss of power and the diesel generator being tagged out, Part 2 is correct.

C. Part 1 is correct, but due to the loss of Div 1 power SWP-MOV55A cannot open.

D. Correct - Under normal conditions, any Standby Service Water Pump can supply the Div 3 DG but under these conditions, SWP-P2A, B, and D are all without power. SWP-P2C which is supplied by the Div 3 diesel is the only available SWP pump. Due to the loss of Div 1 and Div 2 power, SWP-AOV599 will open to supply a return path back to the SBCT since SWP-AOV55A can not open with Div 1 de-energized.

Technical Reference(s): R-STM-0118, Rev 24 Pg 23 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0118 Obj 10a, 13h

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7 Comments:

QUESTION 51 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	264000 A2.06		IR 3.4

Ability to (a) predict the impacts of the following on the Emergency Generators (Diesel/Jet); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
A2.06 Opening normal and/or alternate power to emergency bus.

Proposed Question:

During normal plant operation, the control switch for the Division 1 4160 VAC bus, ENS-ACB06, NORMAL SUPPLY BREAKER is taken to the tripped position.

(1) What is the impact of this condition, and
(2) What actions should be taken as a result?

A. (1) ENS-AC04, ALTERNATE SUPPLY BREAKER, automatically closes to supply the bus.
(2) Transfer Division 2 to its Alternate supply to ensure divisional separation.

B. (1) The bus will remain de-energized.
(2) Depress the Div 1 Diesel Generator Emergency Start pushbutton.

C. (1) The Diesel Generator starts and ties to the bus.
(2) Dispatch an operator to the local diesel control panel to monitor Diesel Generator operation.

D. (1) The Diesel Generator starts and runs unloaded.
(2) Manually load the diesel to supply the bus.

Proposed Answer: C.

Explanation:
A. The alternate supply breaker will not close on undervoltage. The DG will start.
B. The bus will not remain de-energize since the DG will start.
C. Correct - The opening of ENS-ACB06 will cause a bus undervoltage signal which will start the associated diesel generator. The DG output breaker will close to supply the bus when sufficient voltage is established. SOP-0053 directs local monitoring of the diesel generator at the local control panel.
D. The DG starts and loads due to the undervoltage signal generated when the normal supply breaker was opened.

Technical Reference(s): R-STM-0309S Rev 14 Pg 6 of 117 ; SOP-0053 Rev 328 Pg 43 of 115.
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0309S Obj 5a, 6d, 14b
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑2
10 CFR Part 55 Content: 55.41.b.7, b.10
Comments:

QUESTION 52 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	300000 K2.02		IR 3.0

Knowledge of the electrical power supplies to the following:
K2.02 Emergency air compressor

Proposed Question:

The power supply to LSV-C3A, PENETRATION V LEAKAGE CONT AIR COMPRESSOR, is _____________.

A. EHS-MCC2L

B. EHS-MCC2D

C. EHS-MCC8A

D. EHS-MCC8B

Proposed Answer: A.

Explanation:
A. Correct – EHS-MCC2L is a Div 1 safety related power source in the Auxiliary Bldg.
B. EHS-MCC2D is a Div 2 safety related power source. LSV-C3A is a Div 1 component.
C. EHS-MCC8A is a Div 1 safety related power source, but is located in the Control Bldg.
D. EHS-MCC8B is a Div 2 safety related power source and is located in the Control Bldg.

Technical Reference(s): SOP-0034 Rev 13 Pg 34 of 45

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0208 Obj 7b

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 53 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	400000 A4.01		IR 3.1

Ability to manually operate and/or monitor in the control room:
A4.01 CCW indications and control

Proposed Question:

The plant is operating at rated conditions with no equipment out of service. The Reactor Plant Component Cooling Water (CCP) System is in normal operation with CCP-P1A and P1B running. CCP-P1C is in standby.

RPCCW SYSTEM LOW HEADER PRESSURE has alarmed on H13-P870-55. An investigation revealed that the CCP header pressure transmitter CCP-PT127 has failed low. No other alarms or automatic actions have occurred.

What automatic feature failed to function?

A. Trip of the running CRD pump.

B. Start of the standby pump CCP-P1C.

C. Initiation of both Standby Service Water divisions.

D. Isolation of cooling water to the CCP heat exchangers.

Proposed Answer: B.

Explanation:

A. This occurs on either CCP vital loop signal <56 psig. A single transmitter failure will not cause this action.

B. Correct

C. This occurs on when both CCP vital loop signals are <56 psig. A single transmitter failure will not cause this action.

D. Extreme low system pressure will automatically isolate safety and non-safety related portions of SSP, but will not isolate cooling water to the HX's.

Technical Reference(s): R-STM-0115, Rev 6 Pg 9 of 35 ; ARP-P870-55-F04, Rev 14

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115 Obj. 4

Question Source: Bank # RBS-NRC-405

Question History: Last NRC Exam RBS 2004 #53

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41b.7

Comments:

QUESTION 54 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 2	
	K/A #	201001 K2.05		IR 4.5

Knowledge of electrical power supplies to the following:
K2.05 Alternate rod insertion valve solenoids: Plant-Specific

Proposed Question:

The feeder breaker to BYS-PNL02A2 has just tripped. Which of the following valves will be unable to operate due to the loss of its power source?

A. Backup Scram Valves

B. Safety Relief Valves

C. Alternate Rod Insertion valves

D. CMS H2 analyzer sample valves

Proposed Answer: C.

Explanation:
A. The backup scram valves are powered by ENB-PNL02A or ENB-PNL02B.
B. Safety Relief Valves utilizes divisional solenoids powered by ENB-PNL02A or ENB-PNL02B.
C. Correct – ARI valves energized to vent with power supplied from BYS-PNL02A2.
D. CMS-H2 analyzer sample points are supplied by SCM-PNL01A(B).

Technical Reference(s): R-STM-0052 Rev 9 Pg 28 of 68

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0052 Obj 3e

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 55 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	2	Group # 2	
	K/A #	201005 A2.12		IR 3.7

Ability to (a) predict the impacts of the following on the Rod Control and Information System (RCIS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
A2.12 Rod uncoupled: BWR-6

Proposed Question:

RCIS would indicate an uncoupled control rod by ___(1)___ If this were to occur, it would be corrected by utilizing plant procedures to ___(2)___.

A. (1) lighting the DISCONNECTED LED on the rod display module when the ROD UNCOUPLED switch is depressed;
(2) attempt to recouple the rod by driving in to position 46 and withdrawing to position 48.

B. (1) lighting the DISCONNECTED LED on the rod display module when the ROD UNCOUPLED switch is depressed;
(2) isolate the affected control rod to allow it to settle onto the collet fingers.

C. (1) Indicating position 50 on the rod display module when the ROD UNCOUPLED switch is depressed;
(2) attempt to recouple the rod by driving in to position 46 and withdrawing to position 48.

D. (1) indicating position 50 on the rod display module when the ROD UNCOUPLED switch is depressed;
(2) isolate the affected control rod to allow it to settle onto the collet fingers.

Proposed Answer: A.

Explanation:
A. Correct - A rod drop accident occurs when a rod becomes uncouple from its mechanism, sticks in the core and at some later point becomes free and drops out of the core. AOP-0061 requires fully inserting the affected control rod when a control rod drop accident occurs.
B. Part 1 is correct. The rod should be driven in using the INSERT pushbutton, not by isolating/settling the rod.
C. A control rod drift event is the result of stuck collet piston, directional control valve failure, scram valve leakage, or excessive cooling water pressure/flow, but not from being uncoupled. Part 2 is correct.
D. A control rod drift event is the result of stuck collet piston, directional control valve failure, scram valve leakage, or excessive cooling water pressure/flow, but not from being uncoupled. Part 2, See B

Technical Reference(s): ARP-680-07 Rev 33 Page 24 of 51

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPHLO-0549 Obj 4; RLP-STM-0052 Obj 7, 9b, RLP-STM-0500 Obj 3

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

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

Comments: Revised stem, answer, and distractors to match KA; changed technical reference

QUESTION 56 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	2	Group # 2	
	K/A #	202002 A1.08		IR 3.4

Ability to predict and/or monitor changes in parameters associated with operating the Recirculation Flow Control System controls including:
A1.08 Recirculation FCV position

Proposed Question:

While operating at 100% power, a feedwater pump trip resulted in a Recirculation Flow Control Valve (FCV) Runback on both FCVs.

The ATC operator is preparing to reset the runback condition by lowering the output demand of the Recirculation Loop Controllers (B33-K603A(B)) in accordance with the Alarm Response Procedure.

How will the FCV positions be affected by the ATC operator's actions?

FCV position will begin to close ________________________.

A. after % M/A ERROR is zero AND the Cavitation Interlock Reset pushbutton is depressed.

B. after % LIMITER ERROR is zero AND negative deviation is observed on the % SERVO ERROR meter.

C. after % LIMITER ERROR is zero AND the Cavitation Interlock Reset pushbutton is depressed.

D. as soon as the B33-K603A(B) controller is toggled to the left.

Proposed Answer: B.

Explanation:

A. See "B"

B. Correct - A FCV runback signal present applies a 60% drive flow upper limit to FCV position. When the controller is initially toggled to the left, no valve motion will occur. As the control output continues to lower, valve position will begin to close when controller output demand matches the current valve position. At this point a negative servo deviation will be observed and the valve will begin to close. Depressing of the cavitation interlock is only necessary to allow opening the valve, not closing the valve.

C. See "B"

D. See "B"

Technical Reference(s): R-STM-0053 Rev 13 Pg 37-38, 41 of 76; ARP-680-04 Rev 26 Pg 8 of 87

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053 Obj 2c, 2j

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.6 Comments:

QUESTION 57 Rev 1

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 2 Group # 2
K/A # 204000 A4.04 IR 2.8

Ability to manually operate and/or monitor in the control room(Reactor water cleanup):
A4.04 Heat exchanger operation

Proposed Question:

The Reactor Water Cleanup Non-Regenerative Heat Exchangers have experienced a reduction in cooling water flow. If this condition is not corrected, __(1)__ when the Filter Demineralizer inlet temperature reaches the interlock setpoint of __(2)__ .

A. (1) G33-MOVF004, PUMP SUCTION CNMT OUTBOARD ISOL, will isolate
(2) 140°F

B. (1) G33-MOVF001, PUMP SUCTION DRYWELL INBOARD ISOL, will isolate
(2) 140°F

C. (1) G33-MOVF004, PUMP SUCTION CNMT OUTBOARD ISOL, will isolate
(2) 130°F

D. (1) G33-MOVF001, PUMP SUCTION DRYWELL INBOARD ISOL, will isolate
(2) 130°F

Proposed Answer: A.

Explanation:
A. Correct – A reduction in CCP cooling water flow will cause temperature at the outlet of the Non-Regenerative Heat Exchanger (or inlet to the Filter Demineralizer) to rise. When temperature reaches 140°F, G33-MOVF004 will isolate.
B. Although G33-MOVF001 isolates the same penetration as G33-MOVF004, only F004 will receive an isolation signal due to high temperature at the filter demineralizer inlet.
C. At 130°F an alarm will annunciate, but the isolation will not occur until 140°F.
D. At 130°F an alarm will annunciate, but the isolation will not occur until 140°F.

Technical Reference(s): R-STM-0601 Rev 8 Pg 15 of 51 ; AOP-0003, Rev 33 Pg 17 of 22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0601 Obj 3b, 5e, 7b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7

Comments: Revised stem and answer choices to eliminate subset issue

QUESTION 58 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	2	Group # 2		
	K/A #	219000 A3.01			IR 3.3

Ability to monitor automatic operations of the RHR/LPCI: Torus/Suppression Pool Cooling Mode including:
A3.01 Valve operation

Proposed Question:

Residual Heat Removal (RHR) Pump "A" is operating in the suppression pool cooling mode with the following valve alignment:

E12-MOV24A, RHR PUMP A TEST RTN TO SUP PL	OPEN
E12-MOV64A, RHR PUMP A MIN FLOW TO SUP PL	CLOSED
E12-MOV48A, RHR A HX BYPASS VALVE	CLOSED

Which of the following reflects the status of the above valves if a small break LOCA results in receipt of a High Drywell Pressure signal?

	E12-MOV24A	E12-MOV64A	E12-MOV48A
A.	OPEN	CLOSED	OPEN
B.	CLOSED	OPEN	CLOSED
C.	CLOSED	OPEN	OPEN
D.	OPEN	CLOSED	CLOSED

Proposed Answer: C.

Explanation:
A. See "C".
B. See "C".
C. Correct – E12-MOVF024 receives a close signal on High Drywell pressure,
E12-MOV64A will open due to low flow (Test Return and Injection Valve closed),
E12-MOVF048A receives a open signal on High Drywell pressure.
D. See "C".

Technical Reference(s): R-STM-0204 Rev 10 Pg 40, 48 of 63

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204 Obj 6, 9

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.8

Comments: Revised the explanation for answer C to state "E12-MOVF048A receives a open signal on High Drywell pressure

QUESTION 59 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 2	
	K/A #	233000 K4.06		IR 2.9

Knowledge of Fuel Pool Cooling and Clean-Up design feature(s) and/or interlocks which provide for the following:
K4.06 Maintenance of adequate pool level.

Proposed Question:

The design feature of the Fuel Pool Cooling and Cleanup System (SFC) which prevents an uncontrolled loss of upper pool level in the event of a pipe break is/are __________.

A. anti-siphon devices which can NOT be overridden.

B. anti-siphon devices which can be overridden to allow controlled draining of the pool.

C. containment isolation due to low pool level.

D. containment isolation due to high temperatures in SFC equipment areas.

Proposed Answer: B.

Explanation:

A. Anti-siphon device is correct; however, the device may be overridden to allow normal draining of the pools to support vessel disassembly and refueling activities.

B. Correct – The anti-siphon device is simply a hole drilled in the top of SFC piping which serves to break the siphon effect if water level begins lowering due to a pipe break.

C. The Suppression Pool Cooling and Cleanup System isolates due to low level in the upper pool, but the SFC system only isolates on Level 2 or High Drywell Pressure of 1.68 psid.

D. The SFC cooling pump will trip on high room temperature, but containment isolation will only occur on Level 2 or High Drywell Pressure of 1.68 psid.

Technical Reference(s): R-STM-0602 Rev 7 Pg 37 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0602 Obj 2e

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 60 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 2	
	K/A #	256000 K3.02		IR 3.2

Knowledge of the effect that a loss or malfunction of the Reactor Condensate System will have on the following:
K3.02 CRD hydraulics system

Proposed Question:

The plant is operating at 100% power when a malfunction during a Condensate Demineralizer evolution results in a total loss of the Condensate System.

How does this condition affect the Control Rod Drive (CRD) Hydraulics System?

The operating CRD pump will….

A. auto trip and can not be restored.

B. auto trip but can be restored following valve manipulation in the field.

C. remain in service with no change in available Net Positive Suction Head.

D. remain in service with a reduction in available Net Positive Suction Head.

Proposed Answer: D.

Explanation:
A. CRD pump will remain in service with suction from the Condensate Storage Tank (CST).
B. Both the Condensate and CST suction sources are aligned to the CRD pump suction. No valve manipulation is required, and therefore the pump will not trip.
C. The pump will remain in service, however, the loss of Condensate will reduce available NPSH.
D. Correct – The pump will remain in service with a reduction in NPSH due to the loss of one of suction sources.

Technical Reference(s): R-STM-0052 Rev 9, Pg 46 of 68

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0052 Obj 5a, 5b, 15a

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 61 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	2	Group #	2	
	K/A #	271000 K5.07			IR 2.7

Knowledge of the operational implications of the following concepts as they apply to Offgas System:
K5.07 Radioactive decay

Proposed Question:

The first indication of insufficient radioactive decay in the Offgas Charcoal Adsorbers is ___________________.

A. Elevated Post Treat Radiation Levels

B. Elevated Pre Treat Radiation Levels

C. Elevated Area Radiation Levels

D. Elevated Turbine Building Exhaust Radiation Levels

Proposed Answer: A.

Explanation:

A. Correct – The charcoal adsorbers are provided to delay the radioactive gases in the Offgas process flow to allow for sufficient time for radioactive decay to occur. If this does not occur these gases will be released prior decay which will result in elevated radiation levels at the Post Treatment Radiation Monitors.

B. The Pre Treatment Radiation monitor sample point is upstream of the charcoal adsorbers so elevated radiation levels would not be sensed at this point if insufficient radioactive decay was occurring in the charcoal adsorber.

C. If the radioisotope were being released from the adsorbers prior to decay this would not cause elevated area radiation levels

D. The offgas system discharges to the main plant exhaust stack, not the turbine building exhaust stack.

Technical Reference(s): R-STM-0606 Rev 8 Pg 10 of 68

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0606 Obj 3k, 13

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.4

Comments:

QUESTION 62 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 2	
	K/A #	286000 K1.03		IR 2.9

Knowledge of the physical connections and/or cause-effect relationships between Fire Protection System and the following:
K1.03 Reactor water level.

Proposed Question:

In order to inject Fire Water into the reactor in accordance with EOP-0005 Enclosure 7, INJECTION INTO THE RPV WITH FIRE SYSTEM, the Fire Protection System must first be aligned to the ____________________ systems in order to have a flowpath to the reactor vessel.

A. Service Water <u>and</u> Residual Heat Removal "A"

B. Condensate Makeup & Transfer (CNS) <u>and</u> Residual Heat Removal "A"

C. Service Water <u>and</u> Residual Heat Removal "B"

D. Condensate Makeup & Transfer (CNS) <u>and</u> Residual Heat Removal "B"

Proposed Answer: C.

Explanation:

A. Service Water is correct, but there is no interface between SWP and RHR A to provide a path for Fire Protection Water (FPW) to the vessel.

B. Although CNS can be provided to the vessel via RHR A per EOP Enclosure 6, there is no interface to inject FPW to the vessel via CNS and RHR A.

C. Correct – In EOP Enclosure 7, FPW is aligned to the SWP header and then the RHR B flowpath which allows injection of SWP to the RPV is used to provide FPW to the RPV. This lineup is only available in the RHR B loop, not RHR A.

D. Although CNS can be provided to the vessel via RHR B per EOP Enclosure 6, there is no interface to inject FPW to the vessel via CNS and RHR B.

Technical Reference(s): R-STM-0250 Rev 6 Pg 19 of 65 ; EOP-0005, Enclosure 7 Rev 314

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0250 Obj 2; RLP-STM-0204 Obj 13b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.4

Comments:

QUESTION 63 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	2	Group #	2	
	K/A #	288000 G.2.1.30			IR 4.4

Ability to locate and operate components, including local controls. (Plant Ventilation)

Proposed Question:

After associated isolations are bypassed in accordance with EOP-0005 Enclosure 21, to perform Emergency Containment Venting, the operator must manipulate HVR-AOD127, CONTNT PURGE RTN ISOL and HVR-AOD-128, CONTNT RTN INBD ISOL located on control room panel ___(1)___ as well as CPP-MOV105, H2 PURGE FAN DISCH VALVE TO ANNULUS located on ___(2)___.

A. (1) H13-P863; (2) H13-P852 control room back panel

B. (1) H13-P863; (2) CPP-PNL102 on the 171'el of the Aux Bldg.

C. (1) H13-P601; (2) CPP-PNL102 on the 171'el of the Aux Bldg.

D. (1) H13-P601; (2) H13-P852 control room back panel

Proposed Answer: B.

Explanation:

A. Part 1 is correct but CPP-MOV105 control switch is located on CPP-PNL102 in the Aux Bldg. H13-P852 contains isolation logic which must be bypassed to allow opening of HVR air operated dampers which is also performed in this Enclosure.

B. Correct – HVR-AOD127 & 128 control switches are located on H13-P863. CPP-MOV105 control switch is located in the Auxiliary Building 171'el on CPP-PNL102.

C. H13-P601 contains controls for safety related systems, however the controls for HVR-AOD127 & 128 are located on H13-P863. Part 2 is correct.

D. H13-P601 contains controls for safety related systems, however the controls for HVR-AOD127 & 128 are located on H13-P863. CPP-MOV105 control switch is located on CPP-PNL102 in the Aux Bldg. H13-P852 contains isolation logic which must be bypassed to allow opening of HVR air operated dampers which is also performed in this Enclosure.

Technical Reference(s): EOP-0005 Enclosure 21 Rev 314

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0403 Obj 7, 14; RLP-STM-0057 Obj 23

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 64 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 2	
	K/A #	290001 K6.04		IR 3.9

Knowledge of the effect that a loss or malfunction of the following will have on the Secondary Containment:
K6.04 Primary containment system

Proposed Question:

During a Loss of Coolant Accident (LOCA) leakage from the Primary Containment steel structure would be directed into the ___(1)___ where fission products are ___(2)___.

A. (1) Annulus; (2) discharged to the main plant exhaust by annulus pressure control

B. (1) Annulus; (2) filtered and discharged to the main plant exhaust by Standby Gas Treatment

C. (1) Auxiliary building; (2) removed by Emergency containment venting as directed by EOP Enclosures

D. (1) Auxiliary Building; (2) allowed adequate decay time until Normal Containment venting can be performed

Proposed Answer: B.

Explanation:

A. Annulus is correct, but annulus pressure control isolates during a LOCA.

B. Correct-The Annulus(Shield Building) completely surrounds the steel containment structure. If the steel structure were to experience leakage, it would be into this area. This area is maintained at a negative pressure during normal operation by Annulus Pressure Control, but during LOCA conditions, APC isolates and SGTS starts to maintain the annulus pressure negative and filters the air drawn from the annulus prior to sending it to the Main Plant Exhaust Stack.

C. Although the Auxiliary Building is considered part of Secondary Containment, leakage from the steel structure would enter the annulus, not the Auxiliary Building.

D. Although the Auxiliary Building is considered part of Secondary Containment, leakage from the steel structure would enter the annulus, not the Auxiliary Building.

Technical Reference(s): R-STM-0057 Rev 4 Pg 5, of 69; R-STM-0403 Rev 8 Pg 5 of 50

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0057 Obj 29.1; RLP-STM-0403 Obj 12

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑2

10 CFR Part 55 Content: 55.41.b.9

Comments:

QUESTION 65 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 2	
	K/A #	290002 K1.06		IR 3.1

Knowledge of the physical connections and/or cause-effect relationships between Reactor Vessel Internals and the following:
K1.06 HPCS: Plant-Specific

Proposed Question:

The RPV internal component which functions as part of a single system to keep the peak fuel cladding temperature ≤2200°F in the event of a small break LOCA is the ________.

A. High Pressure Core Spray sparger

B. Low Pressure Core Spray sparger

C. Jet Pump Riser

D. Low Pressure Coolant Injection nozzle

Proposed Answer: A.

Explanation:
A. Correct - The HPCS system is designed to maintain PCT ≤ 2200°F during a small break LOCA. The injection point in the vessel is through a spray sparger.
B. Although LPCS has a spray sparger above the core, it can not maintain PCT ≤ 2200°F during a small break LOCA.
C. The jet pumps throat and diffuser are required to maintain 2/3 core coverage during a LOCA, but the riser has no effect in maintaining core coverage during a LOCA.
D. LPCI systems can not maintain PCT ≤ 2200°F during a small break LOCA.

Technical Reference(s): R-STM-0050 Rev 5 Pg 9 of 46; R-STM-0203 Rev 8 Pg 4 of 38

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0050 Obj 3d

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.3

Comments:

QUESTION 66 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3	Group #	Cond of Ops
	K/A #	G.2.1.2		IR 4.1

Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question:

In accordance with EN-OP-115, Conduct of Operations, which of the following is NOT the responsibility of a licensed reactor operator (RO)?

A. Coordinate the activities of shift non-license personnel.

B. Place the plant in a safe condition when faced with unexpected or uncertain conditions

C. Ensure minimum staffing requirements are met at the beginning of each shift

D. Take direction for control manipulations from the person having Control Room Command Function

Proposed Answer: C.

Explanation:
A. Coordination field activities of non-licensed operators is a responsibility of the RO.
B. When faced with uncertain or unexpected plant conditions, the RO has the responsibility to place the plant in a safe condition including power reduction or plant shutdown.
C. Correct - Minimum staffing verification is the responsibility of the Shift Manager.
D. Taking direction for control manipulation from the individual with the Control Room Command Function is an RO responsibility

Technical Reference(s): EN-OP-115 Rev 14 Pg 14 of 89

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0206 Obj 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 67 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3	Group #	Cond of Ops
	K/A #	G.2.1.15		IR 2.7

Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, operations memos, etc

Proposed Question:

The mechanism Operations Managements uses to communicate operational requirements of a continuing nature to the operations crews is ___(1)___, which are to be reviewed by watch standers ___(2)___.

A. (1) night orders; (2) quarterly

B. (1) night orders; (2) each shift

C. (1) standing orders; (2) quarterly

D. (1) standing orders; (2) each shift

Proposed Answer: D.

Explanation:

A. Night orders are used for short duration, not items of a continuing nature. Watch standers are required to review standing orders every shift.

B. Night orders are used for short duration, not items of a continuing nature.

C. Standing orders is correct, but the review must occur every shift. Quarterly applies to the periodic audit performed for continued applicability.

D. Correct - Standing orders can be used by Ops Mgmt to communicate items of continuing nature to Ops crews whereas, night order are for short duration. Watch standers are to review standing orders every shift in eSoms.

Technical Reference(s): EN-OP-112 Rev 1 Pg 5 & 9 of 11

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-402 Obj 3

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 68 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3	Group #	Cond of Ops
	K/A #	G.2.1.44		IR 3.9

Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

Proposed Question:

Select the action required by Technical Specifications if communication between the At the Controls Operator and the refuel platform is lost during fuel handling in the reactor vessel.

A. Verify operability of the refueling interlocks

B. Establish communications with the upper IFTS operator

C. Insert a control rod block

D. Suspend core alterations

Proposed Answer: D.

Explanation:

A. Although refueling interlocks operability is required during fuel movement, the interlocks do not prevent the refueling team from continuing core alterations during a loss of communications.

B. Although the IFTS operator is stationed on the refuel floor, he is not stationed on the platform and therefore establishing communications with the IFTS operator does not meet requirements.

C. Although inserting a rod block prevents the control room from adding positive reactivity to the core, it would not prevent the refueling team from continuing core alterations which is forbidden when communications is lost.

D. Correct - TRM 3.9.11 and FHP-0001 both require immediate suspension of core alts if communications is lost between the main control room and the Refueling Platform personnel during core alterations.

Technical Reference(s): TRM 3.9.11 Condition A. and FHP-0001 Rev 34 Pg 28 of 40

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0055 Obj 9

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 69 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3	Group #	Equip Control
	K/A #	G.2.2.6		IR 3.0

Knowledge of the process for making changes to procedures.

Proposed Question:

In accordance with RBNP-001, DEVELOPMENT AND CONTROL OF RBS PROCEDURES, a COMMENT Procedure Action Request (PAR) may be used for which of the following?

A. To make minor changes that result in a change of intent.

B. To correct typographical errors to a procedure that is currently being implemented.

C. To suggest future improvements or enhancements.

D. To change acceptance criteria.

Proposed Answer: C.

Explanation:
A. A procedure REVISION is required for changes that result in a change of intent.
B. Typographical changes to a procedure being implemented require an EDITORIAL CHANGE.
C. Correct-COMMENT PARs recommend improvements or enhancements to procedures.
D. A change of acceptance criteria also results in a change of intent, so would require a REVISION.

Technical Reference(s): RBNP-001 Rev 35 Pg 19 of 43.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-202 Obj 1

Question Source: Bank # RBS-OPS-1667

Question History: Last NRC Exam RBS NRC April 2010

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 70 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3	Group #	Equip Control
	K/A #	G. 2.2.14		IR 3.9

Knowledge of the process for controlling equipment configuration or status.

Proposed Question:

In accordance with OSP-0038, PROTECTIVE TAGGING GUIDELINES, how are temporary blank flanges which have been installed in a system to support maintenance activities controlled to ensure removal upon system restoration?

The blank flange is ________________.

A. Logged in the Main Control Room Log

B. Logged in the Manipulated Device/Operations Hold Tag Log

C. CAUTION tagged

D. DANGER tagged

Proposed Answer: D.

Explanation:
A. See “D”
B. See “D”
C. See “D”
D. Correct - Per step 3.2.6 of OSP-0038, temporary blank flanges must be danger tagged.

Technical Reference(s): OSP-0038 Rev 35 Pg. 6 of 24

Proposed references to be provided to applicants during examination: None

Learning Objective: R-QC-HLO-RO; R-QC-HLO-SROI; R-QC-HLO-SROU

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 71 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3	Group #	Rad Control
	K/A #	G.2.3.12		IR 3.2

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.

Proposed Question:

Which of the following activities procedurally require Containment Purge operation to control airborne radiological activity in the containment building?

A. Reactor Core Isolation Cooling system manual startup

B. Alternate Decay Heat Removal system startup

C. Suppression Pool Cooling startup

D. Fuel Pool Purification filter backwash

Proposed Answer: A.

Explanation:

A. Correct - SOP-0035 Step 4.2.3 requires Containment Purge operation prior to commencing RCIC manual startup.

B. Although the 3 distractors have the potential to affect radiological conditions in the containment, only RCIC startup procedurally requires operation of containment purge.

C. See “B”

D. See “B”

Technical Reference(s): SOP-0035, Rev 46 Pg 13 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209 Obj 10b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.12

Comments: Revised stem to add the word “procedurally”

QUESTION 72 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3	Group #	Rad Control
	K/A #	G.2.3.14		IR 3.4

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question:

As compared to normal water chemistry, Hydrogen Water Chemistry operation results in ___(1)___ Nitrogen-16 carry-over into the steam lines and ___(2)___ dose rates in steam affected area.

A. (1) greater; (2) lower

B. (1) less; (2) lower

C. (1) greater; (2) higher

D. (1) less; (2) higher

Proposed Answer: C.

Explanation:

A. Part 1 is correct. Part 2 is incorrect. Dose rates are higher due to N16 carryover.

B. Both parts are incorrect.

C. Correct - N16 carry over is greater in an excess H_2 environment due to the formation of NH_3. NH_3 is more likely to carry over with the steam unlike water soluble NO_2 and NO_3 which are formed in an excess O_2 condition. N16 gammas are a significant dose contributor in steam affected areas.

D.Part 1 is incorrect. Part 2 is correct

Technical Reference(s): R-STM-0127 Rev 12 Pg 37 of 68

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0127 Obj 2, 9

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b.12

Comments: Operating Experience: CR-RBS-2002-00195

QUESTION 73 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3	Group #	E-Plan
	K/A #	G.2.4.5		IR 3.7

Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Proposed Question:

Of the procedure types listed below, which one would provide guidance for the notification of state and local agencies in the event of a Fuel Handling accident that resulted in a radioactive release?

A. Fuel Handling Procedures

B. Emergency Implementing Procedures

C. Emergency Operating Procedures

D. Radiation Section Procedures

Proposed Answer: B.

Explanation:

A. Although fuel handling procedures would be in effect, they do not provide guidance for notification of state and local agencies.

B. Correct. The EIPs provide guidance for the classification of events and the subsequent notification to state and local agencies.

C. The EOPs will provide guidance for certain radioactive release events, but do not provide guidance for the notification to state and local agencies.

D. RSPs do not provide guidance regarding notification to state and local agencies during a fuel handling accident resulting in a radioactive release.

Technical Reference(s): EIP-2-006 Rev 40 Pg 2 of 18

Proposed references to be provided to applicants during examination: None

Learning Objective: LEC-EP-022.13 Obj 1 check this

Question Source: Bank # 2008 NRC Exam #75

Question History: Last NRC Exam RBS 2008 NRC Exam

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 74 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐	
	Tier #	3	Group #	E-Plan	
	K/A #	G.2.4.13			IR 4.0

Knowledge of crew roles and responsibilities during EOP usage.

Proposed Question:

During Emergency Operating Procedure (EOP) implementation, which of the following is a responsibility of **BOTH** RO & SRO licensed operators?

A. Answering Decision Steps to determine applicable path.

B. Evaluation of Override Steps to determine when an alternate action is required.

C. Determining when an EOP may be exited.

D. Identification of EOP entry conditions.

Proposed Answer: D.

Explanation:
A. This action is performed by the SRO.
B. This action is performed by the SRO.
C. This action is performed by the SRO.
D. Correct – Identification of EOP entry conditions is the responsibility of all licensed personnel.

Technical Reference(s): OSP-0022 Rev 68 Pg 53 of 77

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0206 Obj 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments: Revised stem to capitalize and bold the word “BOTH” and updated the technical reference.

QUESTION 75 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3	Group #	E-Plan
	K/A #	G.2.4.37		IR 3.0

Knowledge of the lines of authority during implementation of the emergency plan.

Proposed Question:

During an event requiring implementation of the emergency plan, the shift manager assumes the role of ______.

A. Emergency Director

B. EOF (Emergency Operations Facility) Manager

C. OSC (Operations Support Center) Manager

D. Emergency Plant Manager

Proposed Answer: A.

Explanation:

A. Correct – Per EIP-2-002, the Shift Manager assumes the role of ED until properly relieved or until the emergency situation is terminated. The other three positions are manned by the ERO using off-shift personnel.

B. See “A”

C. See “A”

D. See “A”

Technical Reference(s): EIP-2-007 Rev 25 Pg 3 of 9; EIP-2-020 Rev 36 Pg 4 of 58

Proposed references to be provided to applicants during examination: None

Learning Objective: RCBT-EP-NEONCO, Obj

Question Source: Bank RBS 2008 Audit #74

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments: Validation: original question not an RO question; replaced with current question.

QUESTION 76 Rev 2

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑	
	Tier #	1	Group # 1		
	K/A #	295001 G.2.1.23			IR 4.4

Ability to perform specific system and integrated plant procedures during all modes of plant operation. (Partial or Complete Loss of Forced Core Flow Circulation)

Proposed Question:

While reducing core flow, the B FCV malfunctioned and continued to go closed. The ATC took action and shut down the HPU for the B FCV, and the following conditions exist:

- Core Flow 40 mlbm/hr
- Loop A 23 mlbm/hr
- Loop B 17 mlbm/hr
- Reactor Power 57%

Based on these conditions, the CRS would first direct actions in accordance with ____.

A. SOP-0003, Reactor Recirculation System

B. GOP-0004, Single Loop Operation

C. AOP-0024, Thermal Hydraulic Stability Controls

D. Technical Specification 3.4.1, Recirculation Loops Operating

Proposed Answer: C.

Explanation:

A. First actions must be to exit the restricted region using the AOP, then balance flows using the SOP.

B. Single loop operation is not warranted for these conditions.

C. Correct –conditions place the plant in the restricted region of the power to flow map; Insert rods to exit the restricted region.

D. T.S. 3.4.1 requires flow mismatch criteria of ≤ 10% at less than 70% power; current conditions ≈ 7%.

Technical Reference(s): SOP-0003, AOP-0024, TS 3.4.1

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053 Obj 14

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.43.b.2

Comments: Replaced question.

QUESTION 77 Rev 0

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 1	Group # 1	
	K/A # 295006 AA2.01	IR 4.6	

Ability to determine and/or interpret the following as they apply to SCRAM:
AA2.01 Reactor power

Proposed Question:

A reactor scram has just occurred resulting in the following conditions:

• RPV Level	12 inches, rising
• RPV Pressure	1050 psig
• Set Point Set Down	Active
• Control Rod Status	40 rods remain withdrawn
• Reactor Power	3% power

Which of the following represents the appropriate procedure execution for the current conditions?

A. Enter EOP-0001, Transition to EOP-0001A and maintain the Reactor Recirculation Pumps in service.

B. Enter EOP-0001, Transition to EOP-0001A and trip the Reactor Recirculation Pumps.

C. Enter and remain in EOP-0001 and maintain the Reactor Recirculation Pumps in service.

D. Enter and remain in EOP-0001 and trip the Reactor Recirculation Pumps.

Proposed Answer: A.

Explanation:
A. Correct – Based on stem conditions, EOP-0001 is entered (Level 3) and due to control rods remaining withdrawn, transition to EOP-0001A is required. Due to reactor power being <5%, the recirculation pumps are maintain in service due to minimal power reduction effect from securing them at <5% power.
B. It is not appropriate to trip the recirculation pumps during an ATWS with power <5%.
C. Even though power is <5% and there is no EOP-0001 entry condition due to the ATWS, EOP-0001 is entered due to Level 3 and due to control rods remaining withdrawn, transition to EOP-0001A is required.
D. Even though power is <5% and there is no EOP-0001 entry condition due to the ATWS, EOP-0001 is entered due to Level 3 and due to control rods remaining withdrawn, transition to EOP-0001A is required.

Technical Reference(s): EOP-0001 Rev 26; EOP-0001A Rev 26
Proposed references to be provided to applicants during examination: None
Learning Objective: R-LPOPS-HLO-0512 Obj 4, 7
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3
10 CFR Part 55 Content: 55.43.b.5 Comments:

QUESTION 78 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group # 1	
	K/A #	295018 G.2.4.45		IR 4.3

Ability to prioritize and interpret the significance of each annunciator or alarm. (Partial or Complete Loss of Component Cooling Water)

Proposed Question:

The plant is currently in an ATWS condition. The transient began with the receipt of the following alarms:

- REACTOR PLANT COMPNT CLG PUMPS AUTO TRIP
- RPCCW SYSTEM LOW HEADER PRESSURE
- CRD PUMP A OR B AUTO TRIP

Which of the following should the CRS direct to facilitate control rod insertion?

A. ARP-601-22-A1, CRD Pump A or B Auto Trip

B. AOP-0011 Section 5.3, Align SSW to the CRD Pump Bearing Coolers

C. EOP-0005 Enclosure 12, Defeating RPS & ARI Interlocks

D. SOP-0002 Section 5.1, Alternating CRD Pumps

Proposed Answer: B.

Explanation:

A. This procedure would have the operator start the standby CRD pump; this action cannot be performed without CCP header pressure.

B. Correct – Aligning Standby Service Water to the CRD Pumps allows restarting of the system in order to facilitate driving control rods.

C. This Enclosure is normally installed during ATWS conditions to allow draining of the Scram Discharge Volume to allow insertion of another scram signal. In this condition however, with no CRD pump running, the scram can not be reset (hydraulically).

D. This procedure action cannot be performed without CCP header pressure.

Technical Reference(s): AOP-0011 Rev 19 Pg 7 of 8

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP011 Obj 5b, 6b, 6c

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.43.b.5

Comments: In its original form, it could be argued that 3 answers would have worked, though the priority would have been on a single answer.

QUESTION 79 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group # 1	
	K/A #	295016 AA 2.01		IR 4.1

Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT:
AA2.01 Reactor power

Proposed Question:

Given the following conditions:

- The Main Control Room has been evacuated due to a fire
- All immediate actions were attempted prior to leaving the Main Control Room, however the shutdown status of the reactor could not be determined prior to evacuation
- Control of the plant has been established at the Remote Shutdown Panel

Which of the following must the Control Room Supervisor (CRS) use to make the decision to enter EOP-1A, RPV CONTROL – ATWS for these conditions?

A. An estimate of reactor power from calculations using RCIC steam parameters.

B. Determining the status of the Scram Inlet and Outlet Valves on the HCU's.

C. An estimate of reactor power from the number of Safety Relief Valves cycling.

D. Monitoring reactor water level trend versus the number and type of injection systems operating

Proposed Answer: C.

Explanation:

A. There are not enough RCIC parameters available that would make determining reactor power possible.

B. HCU Scram valve status does not provide indication of control rod position and therefore reactor power.

C. Correct – Each SRV passes between 6% and 7% steam flow, which is used to determine the status of the reactor ,following shutdown, decay produces less than 7% steam flow, therefore if only one SRV is cycling the reactor is assumed to be shutdown

D. It would not be possible to determine reactor power based on water level trend and operating injection systems.

Technical Reference(s): RLP-OPS-AOP0031 Rev 2

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP0031 Obj 2

Question Source: Bank RBS-NRC-01154 Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.43.b.5

Comments: Rejected KA; randomly selected current KA and replaced question with a bank question.

QUESTION 80 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group # 1	
	K/A #	295025 G.2.4.21		IR 4.6

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. (High Reactor Pressure)

Proposed Question:

The Technical Specification Bases for the Reactor Recirculation System response to a high reactor pressure signal is to ________________________

A. protect the Reactor Recirculation system from excessive thermal gradient between the steam dome and the recirculation loops.

B. protect the pumps from cavitation due to the decrease in feedwater flow following a reactor scram.

C. insert additional negative reactivity to account for the reduced amount of negative reactivity inserted during the first few feet of control rod travel following a turbine trip at the end of core life.

D. insert negative reactivity during ATWS conditions by increasing voids to counteract the pressure increase maintaining peak RPV pressure below applicable ASME codes.

Proposed Answer: D.

Explanation:

A. The thermal shock interlocks are start permissives and do not trip the Recirc System.

B. This describes the bases for the cavitation interlock transfer for the Recirc Pumps. The caviation interlock is initiated by low feedwater flow, not a high reactor pressure signal.

C. This describes the bases of the End of Cycle Recirc Pump Trip interlocks. EOC-RPT is initiated from a turbine trip, not a high reactor pressure signal.

D. At 1153 psig, the Recirc Pumps trip to off to add negative reactivity by increasing core voiding due to the reduction in core flow. This limits the peak pressure in the RPV to maintain below ASME codes. This setpoint is above the normal pressure scram therefore pressures at this level would be present if a failure to scram would occur.

Technical Reference(s): Technical Specifications Bases 3.3.4.2

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053 Obj 2e, 11

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.2

Comments:

QUESTION 81 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group # 1	
	K/A #	295030 G.2.4.50		IR 4.0

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.
(Low Suppression Pool Water Level)

Proposed Question:

During Mode 1 plant operation, the following annunciators are received:

- H13-P808/83A/F02 SUPPRESSION POOL LEVEL LOW
- H13-P808/84A/F02 SUPPRESSION POOL LEVEL LOW

Corresponding level indicators reads slightly less than 19'6".

Which of the following procedures should the CRS direct for the above conditions?

A. AOP-0003, AUTOMATIC ISOLATIONS AND SOP-0091, Section 6.2 Shutdown of Containment Pool Cooling System Loop A(B) due to SFC isolation.

B. SOP-0035, REACTOR CORE ISOLATION COOLING, Section 5.6 Manual Swap of Suppression Pool and CST Suction Valves to maintain system operability.

C. SOP-0031, HIGH PRESSURE CORE SPRAY, Section 5.3 Manual Swap of Suppression Pool and CST Suction Valves to maintain system operability.

D. AOP-0003, AUTOMATIC ISOLATIONS AND SOP-0140, Section 7.1 Suppression Pool Cooling and Cleanup Shutdown due to SPC isolation.

Proposed Answer: D.

Explanation:

A. Part 1 is correct, but SFC will not isolate. This system selected as a plausible distractor due RHS also isolating on Dryer Storage Pool Level, a component of the SFC system.

B. RCIC LCO entry is not required on SP Level. Additionally, RCIC is normally aligned to the CST. It is not prudent to aligned RCIC to the Sup Pool when it is experiencing low level conditions.

C. HPCS LCO entry is not required on SP Level. Additionally, HPCS is normally aligned to the CST. It is not prudent to aligned HPCS to the Sup Pool when it is experiencing low level conditions.

D. Correct – At 19'6", RHS-AOV62, 63, 64 will isolate tripping the Suppression Pool Cooling and Cleanup System. AOP-0003 is appropriate to verify the isolations occurred and SOP-0104 Section 7.1 is appropriate to place the SPC system in the shutdown lineup.

Technical Reference(s): AOP-0003 Rev 33 Pg 14, 20 if 22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0656 Obj 3d,4b,6,10

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.43.b.5

Comments:

QUESTION 82 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group # 1	
	K/A #	295031 EA 2.03		IR 4.2

Ability to determine and/or interpret the following as they apply to Reactor Low Water Level:
EA2.03 Reactor pressure

Proposed Question:

Plant conditions are as follows:

- Emergency Depressurization is required
- One Reactor feedwater pump is in service
- All injection systems except for Boron, RCIC and CRD are terminated
- 7 ADS SRVs are opened.
- Reactor power is 12%

(1) Which procedure provides guidance for this situation and
(2) When should the CRS direct the commencing of injection into the RPV?

A. (1) EOP-1. (2) When the reactor pressure has lowered to the Minimum Steam Cooling Pressure (MSCP)

B. (1) EOP-1. (2) When reactor pressure has lowered to 46 psig

C. (1) EOP-1A. (2) When the reactor pressure has lowered to the Minimum Steam Cooling Pressure (MSCP)

D. (1) EOP-1A. (2) When reactor pressure has lowered to 46 psig

Proposed Answer: C.

Explanation:
A. See “C”; Part 2 is correct
B. See “C”; 46 psig is the Decay Heat Removal Pressure which is lower than the MSCP (155 psig or greater, depending on the number of SRVs open)
C. Correct-The EOPs require termination of injection prior to ED to prevent the uncontrolled rapid injection of cold water into the vessel. Injection is maintained terminated until adequate core cooling no longer exists. As pressure lowers, ACC is maintained as long as reactor pressure remains above the Minimum Steam Cooling Pressure. At pressures above this value, the steam flow provides sufficient cooling to maintain ACC. Once reactor pressure drops below this value, injection must be restored.
D. See “C”; See B

Technical Reference(s): EOP-0001A Rev 26, RLA-20, RLA-21; EPSTG-2, Rev 16, page 645 of 1117

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-513 Obj 5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.5

Comments: Revised stem, answer, and distractor to make a two part question at the SRO level

QUESTION 83 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group # 2	
	K/A #	295009 G.2.4.6		IR 4.7

Knowledge of EOP mitigation strategies. (Low Reactor Water Level)

Proposed Question:

A Station Blackout has occurred concurrent with an overspeed trip of Reactor Core Isolation Cooling. Current conditions are:

- Level -188", slowly lowering
- Pressure 450 psig

Based on the above conditions, which of the following actions should the CRS take?

A. Enter Steam Cooling section of the EOPs and direct Enclosure 7, INJECTION IN TO THE RPV WITH FIRE SYSTEM

B. Enter Emergency Depressurization section of the EOPs and direct Enclosure 6, INJECTION INTO THE RPV WITH CONDENSATE TRANSFER

C. Enter Steam Cooling section of the EOPs and direct Enclosure 6, INJECTION INTO THE RPV WITH CONDENSATE TRANSFER

D. Enter Emergency Depressurization section of the EOPs and direct Enclosure 7, INJECTION IN TO THE RPV WITH FIRE SYSTEM

Proposed Answer: A.

Explanation:

A. Correct – With no injection available and level below -162" (TAF), Steam Cooling is required. In Steam Cooling, Emergency Depressurization is required at -200". As level slowly lowers from -188" to -200", the fuel is being cooled by steam flow. This time is utilized to align an injection system. Based on initial conditions, the only available injection is Fire Water (FPW) which is aligned per Enclosure 7. Once FPW is aligned the RPV will be depressurized to allow injection from this low pressure system.

B. It would be inappropriate to ED with no injection available while level is still above -200". Condensate Transfer System would not be available under the given conditions.

C. Part 1 is correct, but the Condensate Transfer System would not be available under the given conditions.

D. It would be inappropriate to ED with no injection available while level is still above -200". The fuel has adequate core cooling if maintained above -200". This time should be used to align an injection system.

Technical Reference(s): EOP-0001, Rev 26 ALC-9

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-512 Obj 7

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.43.b.5

Comments:

QUESTION 84 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group # 2	
	K/A #	295020 AA 2.02		IR 3.4

Ability to determine and/or interpret the following as they apply to Inadvertent Containment Isolation:
AA2.02 Drywell/containment temperature

Proposed Question:

Numerous alarms have annunciated in the Main Control Room coincident with the isolation of all Division 2 containment isolation valves. The plant has remained at 100% power. The following conditions exist:

• Div 2 scram solenoids	De-energized
• H13-P680 Div 2 APRM, IRM, SRM status lights	All lit
• H13-P601 CRVICS Div 2 & 3 Status light	Amber light lit
• Drywell temperature	138°F, rising
• Containment temperature	87°F, rising

Which of the following procedures should the CRS direct to address rising temperatures in the containment and drywell?

A. SOP-0059, Containment HVAC System

B. SOP-0060, Drywell Cooling System

C. AOP-0003, Automatic Isolations

D. AOP-0010, Loss of One RPS Bus

Proposed Answer: D.

Explanation:
A. See “D”.
B. See “D”.
C. See “D”.
D. Correct – The given indications are that of an inadvertent containment isolation caused by the loss of the Div 2 RPS bus. The correct action to address this condition is to transfer power to the alternate source, reset the isolation and restore the containment and drywell cooling systems to service. This guidance is provided in AOP-0010, Loss of One RPS Bus.

Technical Reference(s): AOP-0010 Rev 19 Pg 10-11 of 21
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-HLO-0529 Obj 3 & 6
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑2
10 CFR Part 55 Content: 55.43.b.5
Comments:

QUESTION 85 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group # 2	
	K/A #	295034 EA 2.01		IR 4.2

Ability to determine and/or interpret the following as they apply to Secondary Containment Ventilation High Radiation:
EA2.01 Ventilation radiation levels

Proposed Question:

A leak in the Reactor Core Isolation Cooling (RCIC) room has resulted in the following radiation monitor status:

- RMS-RE11A & B, REACTOR BLDG ANNULUS VENT — GREEN
- RMS-RE125, MAIN PLANT EXHAUST DUCT — YELLOW
- RMS-RE110, AUXILIARY BUILDING VENTILATION — RED

Which of the following should the CRS direct based on the current radiation monitor status?

A. AOP-0003, AUTOMATIC ISOLATIONS

B. SOP-0059 Section 5.2, Standby Gas Manual Startup

C. SOP-0065 Section 6.0 Auxiliary Building Ventilation Shutdown

D. OSP-0053 Hard Card for Auxiliary Building Manual Isolation

Proposed Answer: D.

Explanation:

A. There are no automatic actions associated with the one radiation monitor that is in ALARM condition.

B. Although Standby Gas will be started up, this procedure does not perform the requirement to manually isolate the Auxiliary Building.

C. Although the Auxiliary Building Ventilation will be shutdown, this section will not perform the steps required in align SGTS to the annulus.

D. Correct – Unlike most radiation monitors, RMS-RE110 has no associated automatic actuations. EOP-0003, Secondary Containment and Radioactivity Release Control directs manual isolation of the Auxiliary Building as well as alignment of Standby Gas Treatment taking suction from the Auxiliary Building and the Annulus and exhausting to the Main Plant Stack. OSP-0053 states that the OSP-0053 Hard Card is intended to be used when Auxiliary building conditions necessitate building isolation and exhaust filtration provides direction to accomplish these actions.

Technical Reference(s): OSP-0053 Rev 20 Attachment 21

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0409 Obj 8c, 9

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.43.b.5

Comments: Revised Explanation B to state that SOP-0059 does not provide guidance to isolate the auxiliary building and revised Explanation D to contain the OSP-0053 amplifying / guidance for the intended use of the hard card

QUESTION 86 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group # 1	
	K/A #	211000 A2.02		IR 3.4

Ability to (a) predict the impacts of the following on the Standby Liquid Control System; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations:
A2.03 A.C. power failures

Proposed Question:

The plant has experienced an ATWS with the following conditions:

- Reactor power 23%
- Reactor pressure 875 psig
- MSIVs ISOLATED
- Suppression Pool Temp 117°F
- Standby Liquid Control (SLC) Pump 'A' Injecting

Subsequently, a trip of ENS-ACB06, NORMAL SUPPLY BRKR causes a loss of ENS-SWG1A.

Which of the following describes the impact of this condition on SLC and which procedure should the CRS direct in order to mitigate this condition?

A. SLC 'A' will continue to inject. Direct the use of OSP-0053 Attachment 13 Hard card to determine when Hot Shutdown Boron Weight has been injected.

B. SLC 'A' is no longer injecting. Direct the use of OSP-0053 Attachment 13 Hard card to initiate SLC 'B'.

C. SLC 'A' is no longer injecting. Direct the use of Enclosure 15 to commence alternate SLC injection.

D. SLC 'A' will continue to inject. Direct the use of Enclosure 16 for Instrument Air, to allow monitoring SLC Storage Tank Level.

Proposed Answer: B.

Explanation:

A. The power supply to SLC A is EHS-MCC2A which receives power from ENS-SWG1A, so SLC A will be de-energized.

B. Correct – SLC A has lost power due to the loss ENS-SWG1A. OSP-0053 Hard Card provides direction to start the alternate SLC pump "B".

C. The SLC pumps are divisionally separate therefore the loss of ENS-SWG1A will not affect SLC B, so it is available.

D. See A; Instrument air is used for monitoring SLC tank level.

Technical Reference(s): SOP-0028, Rev 17 Pg 16 of 19, EE-001AC EOP-0001A

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0201 Obj 3, 6, 10 RLP-OPS-HLO-513 Obj 5, 6

Question Source: Bank # March 2010 AUDIT # 87

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.43.b.5

Comments: Revised distractor A and B to reference "OSP-0053 Attachment 13 Hard Card" and revised the first part of distractors to not allow RO only knowledge to chose correct answer.

QUESTION 87 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group # 1	
	K/A #	218000 G.2.2.40		IR 4.7

Ability to apply Technical Specifications for a system. (Automatic Depressurization System)

Proposed Question:

7 Days into a High Pressure Core Spray system outage while in Mode 1, it is determined that B21-F051D (Automatic Depressurization System (ADS) valve) is also inoperable.

What are the Tech Spec implications for this condition?

A. Restore the ADS valve to operable status within 7 days or reduce steam dome pressure to < 100 psig in 36 hours.

B. Restore HPCS to operable status within 14 days or be in Mode 3 in 12 hours and mode 4 in 36 hours.

C. Be in Mode 3 in 12 hours and reduce steam dome pressure to <100 psig in 36 hours

D. Within 1 hour initiate action to place the unit in Mode 2 within 7 hours, Mode 3 in 36 hours and Mode 4 within 37 hours.

Proposed Answer: D.

Explanation:
A. The required LCO time for ADS alone would 14 days.
B. This required LCO time for HPCS alone would be 7days remaining.
C. This is the Required Action for multiple ADS valves inoperable. Plausible due to its similarity to LCO 3.0.3
D. Correct – The inoperability of an ADS coincident with HPCS requires entry into LCO 3.0.3.

Technical Reference(s): Technical Specification 3.5.1, 3.0.3

Proposed references to be provided to applicants during examination: TS 3.5.1 No Bases

Learning Objective: RLP-STM-0109 Obj 16; RLP-STM-0203 Obj 14

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑2

10 CFR Part 55 Content: 55.43.b.2

Comments: Revised STEM to state “7 Days into” and revised distractors A and B to reference individual specification for ADS and HPCS.

QUESTION 88 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group # 1	
	K/A #	239002 A2.06		IR 4.3_____

Ability to (a) predict the impacts of the following on the Relief/Safety Valves; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations:
A2.06 Reactor high pressure

Proposed Question:

Following a turbine trip, reactor pressure peaked at 1150 psig resulting in a reactor scram.

(1) How does this condition affect the operation of the Safety Relief Valves and (2) select the procedure the CRS should direct to return the SRVs to pre-transient operation?

A. (1) SRVs will operate with alternate setpoints to reduce pressure loading in containment by preventing multiple actuations in rapid succession. (2) Direct performance of ARP-601-19 Alarm Response Procedure for both Low Low Set logic trip systems.

B. (1) SRVs will operate with alternate setpoints to reduce pressure loading in containment by preventing multiple actuations in rapid succession. (2) Direct performance of SOP-0011 Section 6.3, SVV System Shutdown.

C. (1) SRVs will not operate due to actuation of “Smart Fire” Relay. (2) Direct performance of ARP-601-19 Alarm Response Procedure for both Low Low Set logic trip systems.

D. (1) SRVs will not operate due to actuation of “Smart Fire” Relay (2) Direct performance of SOP-0011 Section 6.3, SVV System Shutdown.

Proposed Answer: A.

Explanation:

A. Correct – Upon actuation of the first SRV pressure trip unit, Low Low Set is activated which lowers the opening setpoint of 2 SRVs and the closing setpoint of 5 SRVs allowing the associated SRVs to open earlier and stay open longer, minimizing the number of cycles which reduced the pressure loading on containment. To return the SRVs to their normal setpoints, the Alarm Response Procedure is utilized to reset the associated logic.

B. Part 1 is correct, but the ARP for the LLS Seal in annunciators must be performed.

C. Smart Fire relay not activated until 1397 psig. Part 2 is correct.

D. Smart Fire relay not activated until 1397 psig. The ARP for LLS Seal in annunciators must be performed.

Technical Reference(s): R-STM-0109 Rev 12 Pg 12-14 of 95; ARP-601-19 Rev 34 Pg 93-94 of 94
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0109 Obj 4c, 5, 27f
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3
10 CFR Part 55 Content: 55.43.b.5 Comments:

QUESTION 89 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group # 1	
	K/A #	262002 A2.02		IR 2.7

Ability to (a) predict the impacts of the following on the Uninterruptable Power Supply (A.C./D.C.); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
A2.02 Over voltage

Proposed Question:

During the performance of STP-302-0102, Power Distribution System Operability Check, the Unit Operator has reported a voltage reading of 123.2 volts AC on VBS-PNL01A. A report from the field confirms 123.2 volts AC at the output of ENB-INV01A. (1) What is the impact of this condition, and (2) what actions should the CRS direct?

A. (1) ENB-INV01A is inoperable and the associated vital bus distribution system is inoperable;
(2) Direct performance of SOP-0048 Section 5.5, Transfer from ENB-INV01A to ENB-INV01A1 supplying VBS-PNL01A.

B. (1) Only ENB-INV01A is inoperable.
(2) Direct performance of SOP-0048 Section 5.5, Transfer from ENB-INV01A to ENB-INV01A1 supplying VBS-PNL01A.

C. (1) Only ENB-INV01A is inoperable
(2) Direct performance of SOP-0048 Section 5.2 Transferring an ENB Inverter from Normal Operation to Maintenance Bypass.

D. (1) ENB-INV01A is inoperable and the associated vital bus distribution system is inoperable.
(2) Direct performance of SOP-0048 Section 5.2 Transferring an ENB Inverter from Normal Operation to Maintenance Bypass.

Proposed Answer: A.

Explanation:

A. Correct - The surveillance contains acceptance criteria for both the inverter and the vital bus. Both are outside of the given acceptance criteria therefore the surveillance requirement is not met so both components must be declared inop (TS 3.8.7 & 3.8.9). Transferring to ENB-INV01A will satisfy TS 3.8.7 and will correct the voltage condition which will also allow exit of TS 3.8.9).

B. The vital bus distribution system is also inop due to voltage being outside the acceptance criteria. Part 2 is correct.

C. Part 1 is incorrect, see "B". Operating an inverter in Manual Bypass does not satisfy TS 3.8.7. The correct action is to transfer to the alternate inverter.

D. Part 1 is correct, but operating an inverter in Manual Bypass does not satisfy TS 3.8.7. The correct action is to transfer to the alternate inverter.

Technical Reference(s): STP-302-0102, TS 3.8.7, TS 3.8.9
Proposed references to be provided to applicants during examination: STP-302-0102, TS 3.8.7, TS 3.8.9
Learning Objective: RLP-STM-0300 Obj 10
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑4
10 CFR Part 55 Content: 55.43.b.5
Comments: Removed Bases from Proposed References to be provided.

QUESTION 90 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑	
	Tier #	2	Group #	1	
	K/A #	400000 G.2.4.11			IR 4.2

Knowledge of abnormal condition procedures. (Component Cooling Water System)

Proposed Question:

A total loss of Reactor Plant Component Cooling Water has occurred. In order to accomplish the direction of AOP-0011, LOSS OF REACTOR PLANT COMPONENT COOLING WATER regarding operation of the Reactor Recirculation System, the CRS should direct performance of which of the following procedures?

A. OSP-0053 Attachment 18, Securing Reactor Recirculation Pumps, ONLY

B. OSP-0053 Attachment 19, Isolation of Reactor Recirculation Pumps, ONLY

C. OSP-0053 Attachment 18 AND OSP-0053 Attachment 19

D. SOP-0003 Section 5.8. Restoring Seal Purge to an Operating Recirculation Pump ONLY

Proposed Answer: C.

Explanation:
A. This Attachment is performed, but Attachment 19 is also required.
B. This Attachment is performed, but Attachment 18 is also required.
C. Correct – Both Attachments 18 and 19 are required. With a loss of cooling only tripping of the pumps is required. With loss of cooling and seal purge both tripping and isolating the pumps is required per AOP-0011. Attachments 18 and 19 are both required to accomplish this action.
D. With a loss of CCP, seal purge cannot be restored due to CRD tripping on low CCP pressure. In addition, the recirc pumps will not remain in service due to the loss of cooling, so there would be no “Operating Recirculation Pump”.

Technical Reference(s): AOP-00011 Rev 19 Pg 5 of 8

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115 Obj 9a,11a; RLP-OPS-AOP0011 Obj 4,6b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.43.b.5

Comments: Revised stem and distractor D as recommended by NRC comments

QUESTION 91 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group # 2	
	K/A #	201003 A2.09		IR3.4

Ability to (a) predict the impacts of Low reactor pressure **on the Control Rod and Drive Mechanism; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations:**

Proposed Question:

During a plant startup an accumulator fault is received on control rod 28-29 due to low accumulator pressure.

Reactor Pressure	530 psig
Charging water header pressure	1750 psig

What is the impact of this condition and what action should the CRS direct to mitigate the condition?

A. (1) The control rod may not fully insert on a scram signal due to insufficient pressure from the accumulator and/or the reactor via the ball check valve.
(2) SOP-0071, ROD CONTROL AND INFORMATION SYSTEM, Section 5.3 Continuous insert using In Timer Skip to fully insert the control rod.

B. (1) The control rod may not fully insert on a scram signal due to insufficient pressure from the accumulator and/or the reactor via the ball check valve.
(2) SOP-0002, CONTROL ROD DRIVE HYDRAULICS, Section 4.3 Charging HCU Accumulators with Nitrogen, to restore accumulator pressure.

C. (1) The control rod may drift due to higher differential pressure across the drive piston.
(2) SOP-0071, ROD CONTROL AND INFORMATION SYSTEM, Section 5.3 Continuous insert using In Timer Skip to fully insert the control rod.

D. (1) The control rod may drift due to higher differential pressure across the drive piston.
(2) SOP-0002, CONTROL ROD DRIVE HYDRAULICS, Section 4.3 Charging HCU Accumulators with Nitrogen, to restore accumulator pressure.

Proposed Answer: B.

Explanation:

A. Part 1 is correct, but rod insertion is not required since charging water header pressure is >1540 psig.

B. Correct – The control rod is capable of scramming by either accumulator or reactor pressure via the ball check valve on the CRD mechanism. With low pressure in the reactor and the accumulator the rod may not fully scram when required. Since charging water header pressure is >1540 psig, the rod doesn't have to be inserted. The rod must, however, be declared inop within 1 hour. It is prudent therefore to charge the associated accumulator to exit the condition.

C. Both parts incorrect. The differential pressure across the drive piston is maintained by the pressure control valve which adjusts as reactor pressure changes. See "B" for explanation of Part 2.

D. Part 1 is incorrect. Part 2 is correct.

Technical Reference(s): Tech Spec 3.1.5 and bases; ARP-680-07-C03 Rev 33 Pg 25 of 51

Proposed references to be provided to applicants during examination: TS 3.1.5

Learning Objective: RLP-STM-0052 Obj 8q, 11b, 12

Question Source:	New	Question History:	Last NRC Exam None
Cognitive Level:	Memory or Fundamental Knowledge ☐		Comprehension or Analysis ☑3
10 CFR Part 55 Content:	55.43.b.5	Comments:	

QUESTION 92 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group # 2	
	K/A #	223001 A2.06		IR 4.1

Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
A2.06 High containment pressure: Mark-III

Proposed Question:

A steam leak has occurred resulting in the following plant conditions:

- Drywell Pressure 1.4 psid
- Containment Pressure 2.1 psig
- Normal containment vent and purge is in service

(1) Which of the following describes the impact of the above conditions; and (2) what procedure should the CRS direct to mitigate the consequences of this condition?

A. (1) Damage to the HVAC ductwork;
(2) SOP-0059, Containment HVAC System, to secure normal containment vent and purge.

B. (1) Damage to the HVAC ductwork;
(2) EOP Enclosure 21 for Emergency Containment Venting.

C. (1) Containment vent valves cannot be opened;
(2) SOP-0059, Containment HVAC System, to secure normal containment vent and purge.

D. (1) Containment vent valves cannot be opened;
(2) EOP Enclosure 21 for Emergency Containment Venting.

Proposed Answer: A.

Explanation:
A. Correct – Above 2 psig, normal containment vent and purge must be secured in order to prevent damage to the Aux Bldg HVAC ductwork which is used during this evolution.
B. Part 1 is correct, but Enclosure 21 is used to maintain pressure below 30 psig. It would be inappropriate to override interlocks and vent containment at 2.1 psig.
C. The pressure which corresponds to the inability to operate containment vent valves for venting purposes is 15 psig. Part 2 is correct.
D. See "C". Enclosure 21 is used to maintain pressure below 30 psig. It would be inappropriate to override interlocks and vent containment at 2.1 psig.
Technical Reference(s): EOP-0002, EPSTG*0002 B-8-11
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-HLO-0514 Obj 5 & 6
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3
10 CFR Part 55 Content: 55.43.b.5
Comments: Added procedure titles ; Changed question #11

QUESTION 93 Rev 2

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group # 2	
	K/A #	259001 G.2.4.46		IR 4.2

Ability to verify that the alarms are consistent with the plant conditions. (Reactor Feedwater System)

Proposed Question:

Power ascension is in progress per GOP-0001, PLANT STARTUP when the following annunciator is received:

- REACTOR HIGH/LOW WATER LEVEL

The ATC operator reports the following:

- C33-R606 'A' Narrow Range Level 37 inches
- C33-R606 'B' Narrow Range Level 29 inches
- C33-R606 'C' Narrow Range Level 36 inches
- Startup Feedwater Regulating Valve Manual, closed
- "A" Feedwater Regulating Valve Auto, 90% open
- "B" Feedwater Regulating Valve Manual, closed
- "C" Feedwater Regulating Valve Manual, closed
- C33-R606 'B' Narrow Range Level Selected

Which of the following procedures should the CRS implement first to alleviate the condition?

A. AOP-0006, Condensate/Feedwater Failures

B. AOP-0042, Loss of Instrument Bus

C. SOP-0001, Nuclear Boiler Instrumentation

D. SOP-0009, Reactor Feedwater System

Proposed Answer: A.

Explanation:

A. Correct – With C33-R606 B level selected, and the B level reading 29", the Feed Reg valve should be full open. Indication is that the valve is only 90% open, indicating a non-responding FRV. AOP-0006, directs lowering power to within the capacity of the feed and condensate systems.

B. Although a loss of VBN-PNL01B1 would cause FRVs to lock up it would also cause A narrow range indication to fail low.

C. SOP-0001 has guidance for venting a level instrument, it has no guidance for a stuck FRV.

D. SOP-0009 has direction to place another FRV in service, which would be warranted here, but only after entering AOP-0006 and in accordance with direction in AOP-0006.

Technical Reference(s): SOP-0009, Rev 62 Pg 34, 35 of 99

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107A Obj 6, 7, 9; RLP-STM-0107B Obj 5, 9, 11,

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.43.b.5

Comments: Revised to SRO level question, application of procedural guidance.

QUESTION 94 Rev 1

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 3	Group #	Cond. of Ops
	K/A # G.2.1.4		IR 3.8

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc

Proposed Question:

In accordance with the Technical Specification manning requirement, Shift crew composition may be _____ less than the minimum requirements for a period of time not to exceed ______ hours in order to accommodate unexpected absence of on-duty shift crew members.

A. Two (2), One (1)

B. One (1), One (1)

C. Two (2), Two (2)

D. One (1), Two (2)

Proposed Answer: D.

Explanation:
A. See "D"
B. See "D"
C. See "D"
D. Tech Spec 5.2.2.c. states the minimum staffing requirements may be one less than minimum requirements (2 ROs, 2 SROs, 1 STA) for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

Technical Reference(s): Technical Specification 5.2.2.c.
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-HLO-0416 Obj T2
Question Source: Bank # RBS-NRC-641
Question History: Last NRC Exam RBS Feb 1999
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3
10 CFR Part 55 Content: 55.43.b.1
Comments: Revised stem and distractors to match the suggestion from the NRC

QUESTION 95 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3	Group #	Cond. of Ops
	K/A #	G.2.1.41		IR 3.7

Knowledge of the refueling process.

Proposed Question:

Prior to entering Mode 5, the list of applicable Technical Specifications in _________________ must be verified met.

A. REP-0029, Fuel Movement Plan

B. FHP-0001, Control of Fuel Handling and Refuel Operations

C. GOP-0002, Plant Shutdown

D. GMP-0102, Reactor Vessel Disassembly

Proposed Answer: B.

Explanation:

A. REP-0029 contain guidance regarding assembly of the fuel movement plan and other fuel movement rules, but does not contain the list of applicable Mode 5 Tech Specs.

B. Correct – Attachment 2 of FHP-0001 contains applicable Mode 5 Tech Specs which must be met prior to entry into Mode 5.

C. GOP-0002 provides guidance for plant shutdown and entry into Mode 4, but does not contain guidance concerning applicable Mode 5 Tech Specs.

D. GMP-0102 contains the direction to notify the control room prior to detensioning the vessel head which results in entry into Mode 5, but does not provide the list of applicable Tech Specs which must be met prior to entry into Mode 5.

Technical Reference(s): FHP-0001 Rev 34 Pg 34 of 40

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-RF-PROC Obj 28

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.7

Comments:

QUESTION 96 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3	Group #	Equip Control
	K/A #	G.2.2.7		IR 3.6

Knowledge of the process for conducting special or infrequent tests.

Proposed Question:

In accordance with EN-OP-116, INFREQUENTLY PERFORMED TESTS OR EVOLUTIONS, the individual who has the experience and expertise to exercise “continuous responsibility” for the oversight of a particular test or evolution is the _______(1)_________ who is designated by the ________(2)_______.

A. (1) Senior Line Manger (SLM); (2) General Manger Plant Operations (GMPO)

B. (1) Control Room Supervisor (CRS); (2) Operations Shift Manager (OSM)

C. (1) GMPO; (2) Site Vice President

D. (1) Test Coordinator; (2) OSM

Proposed Answer: A.

Explanation:
A. The SLM who is senior to the Shift Manager has continuous responsibility for the oversight of the IPTE and is assigned by the GMPO.
B. See “A”.
C. See “A”.
D. The Test Coordinator is responsible for the overall conduct of the test, but not the oversight (see “A”)

Technical Reference(s): EN-OP-116 Rev 12, Pg 7 of 35

Proposed references to be provided to applicants during examination: None

Learning Objective: Training provided by R-QC-HLO-SROU & R-QC-HLO-SROI

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.3

Comments:

QUESTION 97 Rev 2

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3	Group #	Equip Control
	K/A #	G.2.2.20		IR 3.8

Knowledge of the process for managing troubleshooting activities.

Proposed Question:

A component failure requires investigation to determine the specific type of repair required to return the component to operation.

In accordance with EN-MA-125, Troubleshooting Control of Maintenance Activities, who is responsible for determining the "Troubleshooting Determination Level?"

A. Operations Manager, <u>OR</u> designee

B. Maintenance Manager <u>AND</u> Operations Manager

C. FIN Superintendent <u>OR</u> Shift Manager

D. Responsible Maintenance Supervisor <u>AND</u> Shift Manager

Proposed Answer: D.

Explanation:

A. Ops Manager may be required to approve a troubleshooting plan, but is not required to perform the troubleshooting determination level.

B. See A.

C. Even if the FIN superintendent were the RMS designee, the procedure requires **both** not either.

D. Correct – Section 5.3, Troubleshooting Determination Level, requires both the RMS and the SM to determine the determination level.

Technical Reference(s): EN-MA-125 Rev 16 Pg 18 of 54

Proposed references to be provided to applicants during examination: None

Learning Objective: Training provided by R-QC-HLO-SROU & R-QC-HLO-SROI

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑4 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.5

Comments: Replaced the question after submittal.

QUESTION 98 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3	Group #	Rad Control
	K/A #	G.2.3.11		IR 4.3

Ability to control radiation releases.

Proposed Question:

Which of the following is required to discharge an LWS tank to the Mississippi River if RMS-RE107 is INOPERABLE?

A. Two independent samples of the tank are analyzed. One qualified member of the Chemistry staff and one qualified member of the Radwaste staff independently verify the release rate calculations and the discharge valve lineup.

B. A single sample is analyzed by two qualified members of the Chemistry staff independently. Two qualified members of the Radwaste staff independently verify the discharge valve lineup.

C. Two independent samples of the tank are analyzed. Two qualified members of the Chemistry staff independently verify the release rate calculations. Two qualified members of the Radwaste staff independently verify the discharge valve lineup.

D. A single sample of the tank is analyzed. One qualified member of the Chemistry staff verifies the release rate calculation and one qualified member of the Radwaste staff verifies the discharge valve lineup.

Proposed Answer: C.

Explanation:
A. See C.
B. See C.
C. Correct – ADM-0054 Section 5.5 requires 2 independent samples, 2 Chemistry technicians to verify calculations and 2 qualified members of the Radwaste staff to verify the discharge valve lineup. These actions satisfy the requirements of TRM 3.3.11.2. This is a short completion time LCO, in that it is required prior to discharge. This requirement should be committed to memory.
D. See C.

Technical Reference(s): TRM 3.3.11.2 & ADM-0054 Rev 6A Section 5.5 Pg 9 of 12.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0603 Obj 8c

Question Source: Bank # 2008 RBS NRC #97

Question History: Last NRC Exam November 2012

Cognitive Level: Memory or Fundamental Knowledge ☑3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43b.2 & b.4

Comments: Appeared on one of last 2 NRC exams (1 of 2)

QUESTION 99 Rev 01

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 3	Group #	E-Plan
	K/A # G.2.4.22		IR 4.4

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Proposed Question:

In accordance with 10CFR50.54x, a licensee may take action that departs from a license condition or a technical specification under certain conditions when this action is needed to ______________________________.

A. extend public exposure limits in the event of an emergency.

B. protect public health and safety.

C. prevent a release rate in excess of federally approved operating limits.

D. extend safety related equipment out of service time.

Proposed Answer: B.

Explanation:
A. See "B".
B. Correct – The departure from license condition is only allowed during emergency conditions to protect public health and safety.
C. See "B".
D. See "B".

Technical Reference(s): 10CFR50.54x, EN-OP-115 Rev 14 Pg 5 of 89

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0206 Obj 16

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.1

Comments: changed distractors A and C to address credibility issues

QUESTION 100 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3	Group #	E-Plan
	K/A #	G.2.4.42		IR 3.8

Knowledge of emergency response facilities.

Proposed Question:

After all emergency response facilities are operational during a declared plant emergency, personnel in which of the areas below have the overall responsibility for determining recommended protective measures for offsite persons within the Emergency Planning Zone and for communication of these recommendations?

A. Main Control Room

B. Operations Support Center

C. Technical Support Center

D. Emergency Operating Facility

Proposed Answer: D.

Explanation:
A. See “D”.
B. See “D”.
C. See “D”.
D. The shift manager in the MCR has this responsibility initially, but after all facilities are manned, this responsibility is transferred to the Emergency Director in the Emergency Operations Facility.

Technical Reference(s): EIP-2-007 Rev 25 Pg 3 of 9

Proposed references to be provided to applicants during examination: None

Learning Objective: RCBT-EP-SRORMED Obj 17

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43b.5

Comments:

REFERENCE HANDOUTS

CAUTIONS

CAUTION #1

RPV water level indications are affected by instrument run temperatures and RPV pressure:

1. If the temp near **any** instrument run is above the RPVST (Fig. 1), the instrument may be unreliable due to boiling in the run

   Max DW Temp at El. 145 ft, 440°F

   Max CTMT Temp at El. 119 ft, 200°F

2. Each of the following instruments may be used to determine RPV water level only when the instrument reads above the Minimum Indicated Level associated with the highest temperature near the instrument reference leg vertical run.

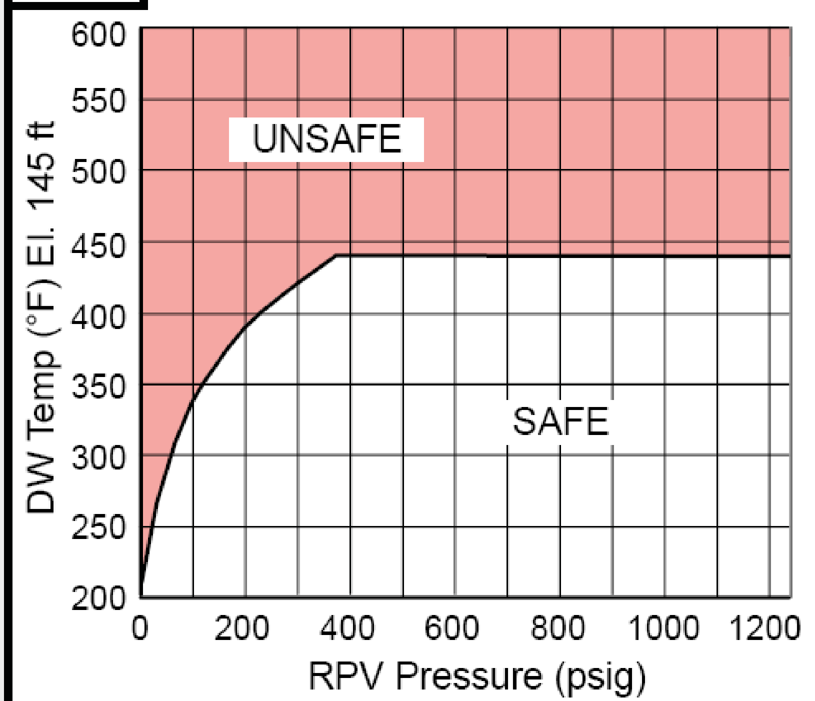
Containment Temperature (°F) El. 119 ft

Containment Temperature (°F)	Narrow	Wide	Fuel Zone
100–200	1 in.	-142 in.	-310 in.
70–100	0 in.	-159 in.	-310 in.

Drywell Temperature (°F) El. 145 ft

Drywell Temperature (°F)	Upset	Shutdown
300–440	53 in.	55 in.
200–300	20 in.	29 in.
100–200	2 in.	16 in.

1 RPV SATURATION TEMPERATURE RPVST



INITIATING STANDBY LIQUID CONTROL

STANDBY LIQUID CONTROL INJECTION REQUIREMENTS

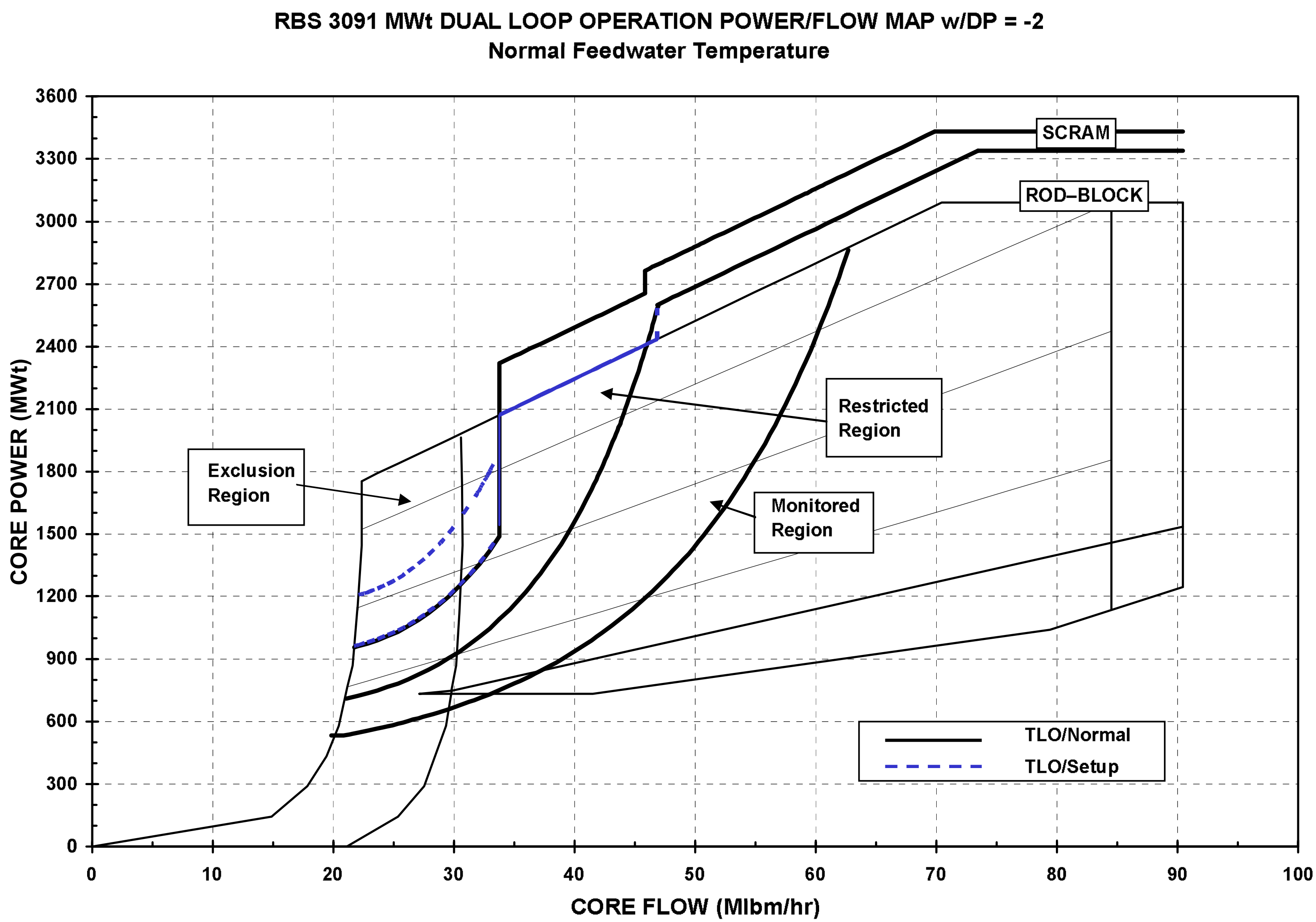
TANK LEVEL PRIOR TO INJECTION GAL	TANK LEVEL AFTER INJECTION OF 69 lb Boron (approximately 16 min inj time) GAL	TANK LEVEL AFTER INJECTION OF 166 lb Boron (approximately 38 min inj time) GAL

NOTE

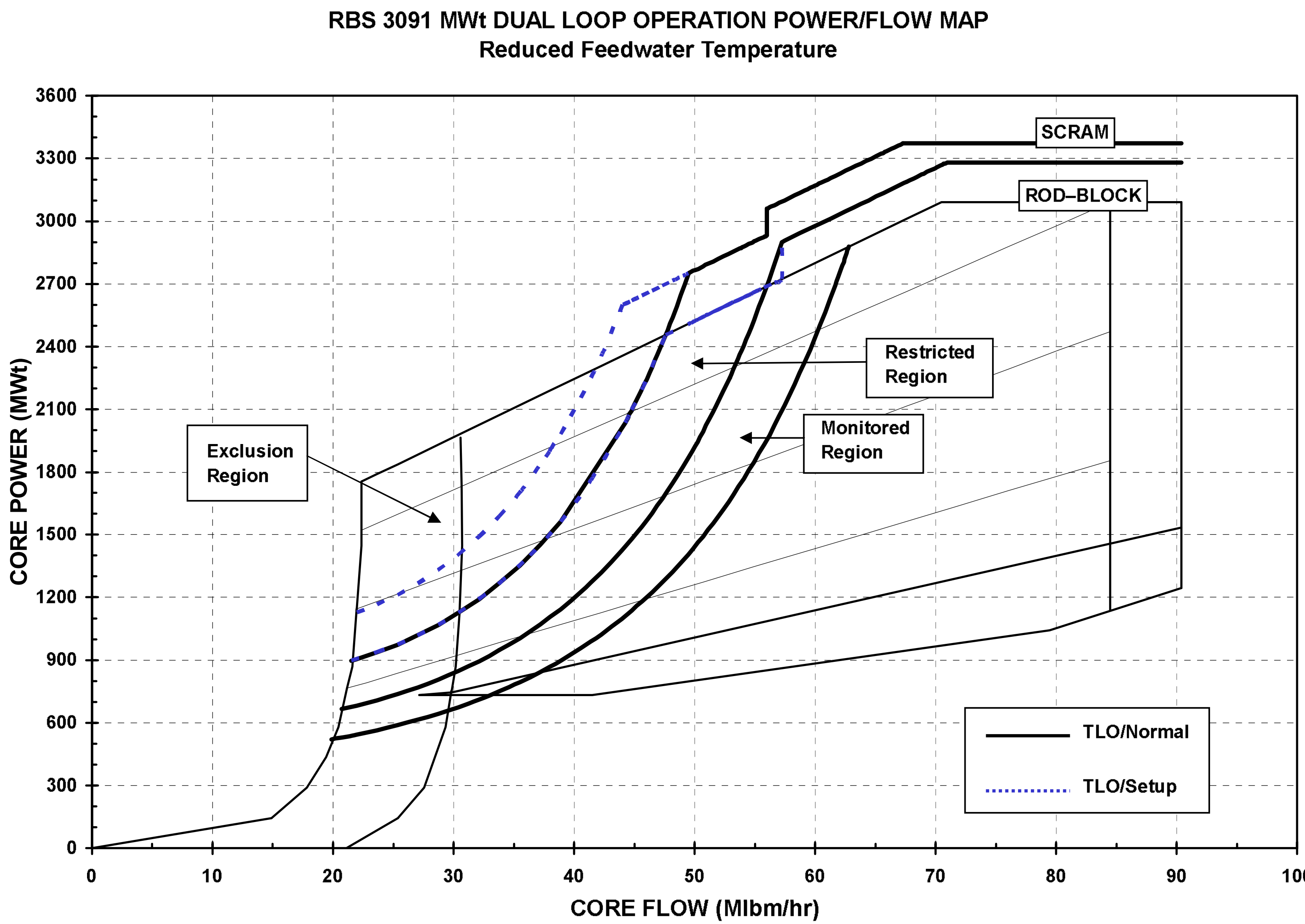
WHEN tank level falls between values, THEN the smaller value should be used.

1531	905	0
1550	924	19
1600	974	69
1700	1074	169
1800	1174	269
1900	1274	369
2000	1374	469
2100	1474	569
2200	1574	669
2300	1674	769
2400	1774	869
2500	1874	969
2600	1974	1069
2700	2074	1169
2800	2174	1269
2900	2274	1369
3000	2374	1469
3100	2474	1569
3200	2574	1669
3300	2674	1769
3400	2774	1869
3500	2874	1969
3600	2974	2069
3700	3074	2169
3800	3174	2269
3900	3274	2369
4000	3374	2469
4100	3474	2569
4200	3574	2669

RBS DUAL LOOP OPERATION POWER/FLOW MAPS

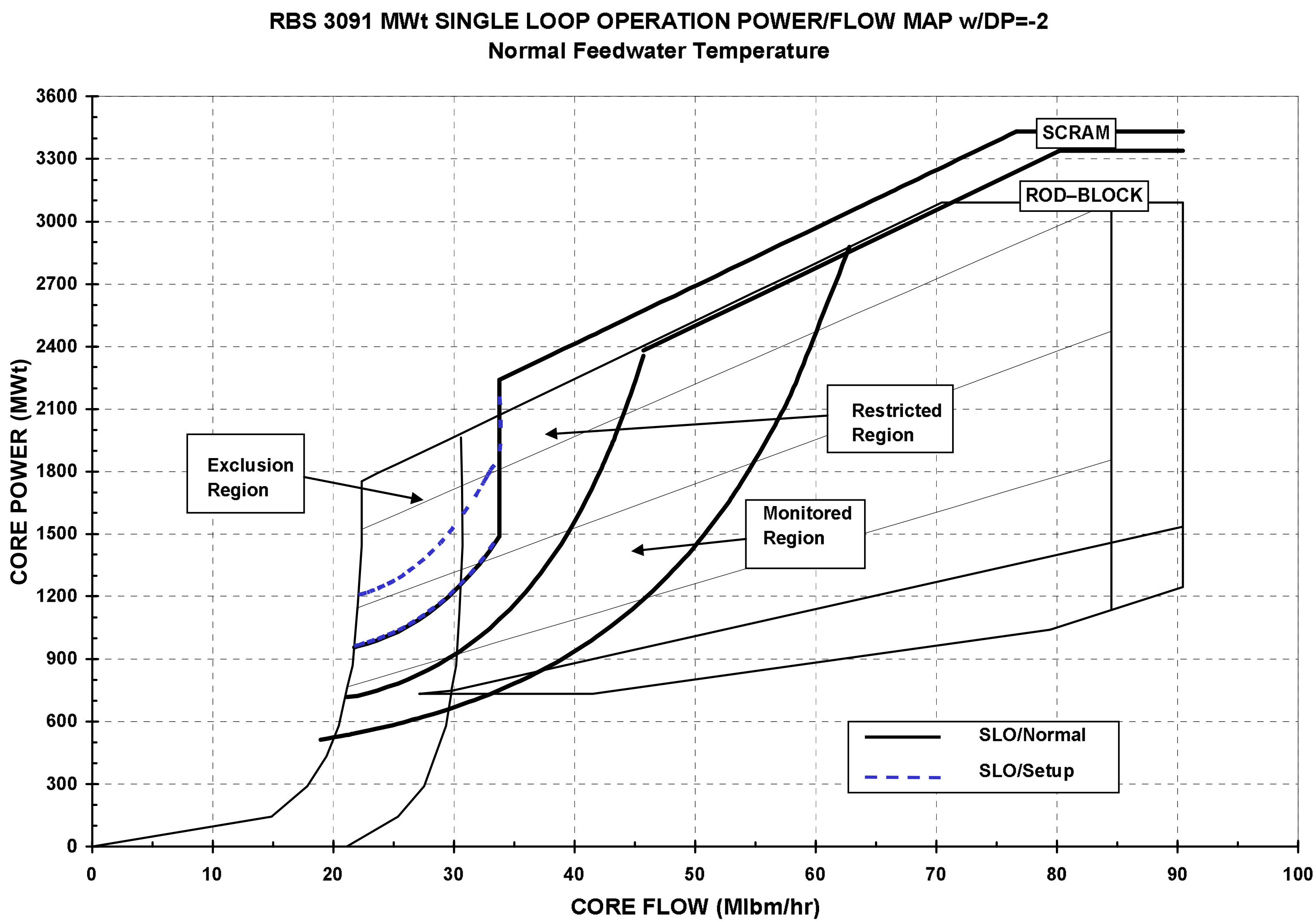


RBS DUAL LOOP OPERATION POWER/FLOW MAPS

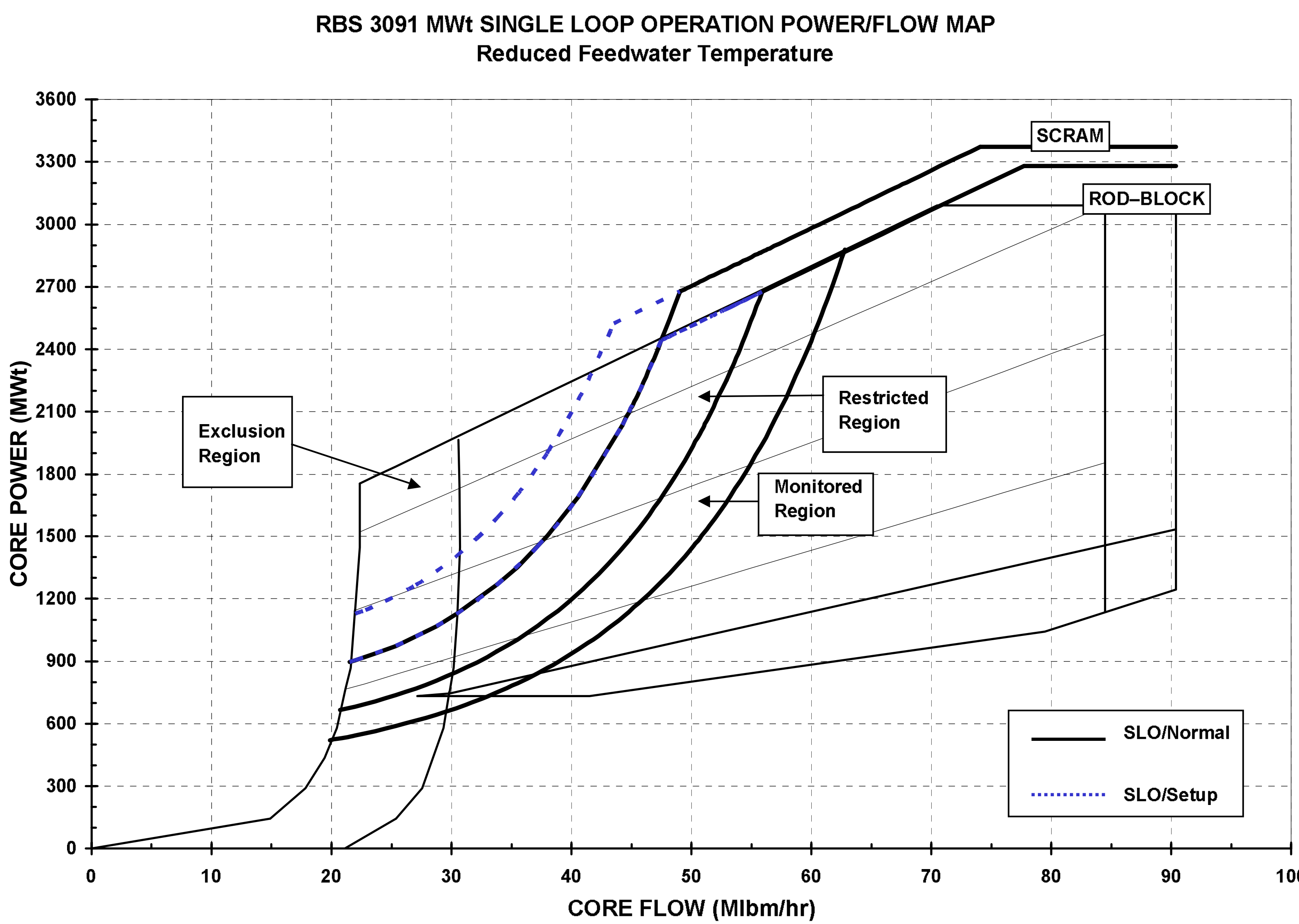


(Used when below the - 50°F Curve of AOP-0007, Loss of Feedwater Heating)

RBS SINGLE LOOP OPERATION POWER/FLOW MAPS



RBS SINGLE LOOP OPERATION POWER/FLOW MAPS



(Used when below the - 50°F Curve of AOP-0007, Loss of Feedwater Heating)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS -Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of seven safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 100 psig.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable to HPCS.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC is required to be OPERABLE. AND B.2 Restore HPCS System to OPERABLE status.	1 hour 14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two ECCS injection subsystems inoperable. <u>OR</u> One ECCS injection and one ECCS spray subsystem inoperable.	C.1 Restore one ECCS injection/spray subsystem to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	12 hours 36 hours
E. One ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days
F. One ADS valve inoperable. <u>AND</u> One low pressure ECCS injection/spray subsystem inoperable.	F.1 Restore ADS valve to OPERABLE status. <u>OR</u> F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
G. Two or more ADS valves inoperable. <u>OR</u>	G.1 Be in MODE 3. <u>AND</u>	12 hours (continued)

ACTIONS

CONDTION	REQUIRED ACTION	COMPLETION TIME
G. (continued) Required Action and associated Completion Time of Condition E or F not met.	G.2 Reduce reactor steam dome pressure to $\leq$ 100 psig.	36 hours
H. HPCS and Low Pressure Core Spray (LPCS) Systems inoperable. <u>OR</u> Three or more ECCS injection/spray subsystems inoperable. <u>OR</u> HPCS System and one or more ADS valves inoperable. <u>OR</u> Two or more ECCS injection/spray subsystems and one or more ADS valves inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.1.2	----------------------------NOTE--------------------------------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the residual heat removal cut in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. --	
	Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.1.3	Verify ADS accumulator supply pressure is $\geq$ 131 psig.	31 days
SR 3.5.1.4	Verify each ECCS pump develops the specified flow rate with the specified pump differential pressure. SYSTEM — FLOW RATE — PUMP DIFFERENTIAL PRESSURE LPCS — $\geq$ 5010 gpm — $\geq$ 282 psid LPCI — $\geq$ 5050 gpm — $\geq$ 102 psid HPCS — $\geq$ 5010 gpm — $\geq$ 415 psid	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.1.5	----------------------------NOTE------------------------------- Vessel injection/spray may be excluded. -- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.6	------------------------------NOTE----------------------------- Valve actuation may be excluded. -- Verify the ADS actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.7	-----------------------------NOTE------------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each ADS valve relief mode actuator strokes when manually actuated.	In accordance with the Inservice Testing Program on a STAGGERED TEST BASIS for each valve solenoid
SR 3.5.1.8	------------------------------NOTE----------------------------- ECCS actuation instrumentation is excluded. -- Verify the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is within limits.	24 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters–Operating

LCO 3.8.7 The Division I and Division II inverters shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems–Operating," with any AC vital bus de-energized.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Division I or II inverter inoperable.	A.1 Restore Division I and II inverters to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to required AC vital buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems–Operating

LCO 3.8.9 Division I, Division II, and Division III AC and DC, and Division I and II AC vital bus electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

--NOTE--
Division III electrical power distribution subsystems are not required to be OPERABLE when High Pressure Core Spray System and Standby Service Water pump 2C are inoperable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Division I or II AC electrical power distribution subsystems inoperable.	A.1 Restore Division I and II AC electrical power distribution subsystems to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
B. One or more Division I or II AC vital bus distribution subsystems inoperable.	B.1 Restore Division I and II AC vital bus distribution subsystems to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO

(continued)

ACTION (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Division I or II DC electrical power distribution subsystems inoperable.	C.1 Restore Division I and II DC electrical power distribution subsystems to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	12 hours 36 hours
E. One or more Division III AC or DC electrical power distribution subsystems inoperable.	E.1 Declare High Pressure Core Spray System and Standby Service Water System pump 2C inoperable.	Immediately
F. Two or more divisions with inoperable distribution subsystems that result in a loss of function.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	7 days

*G12.1.15

ENTERGY

RIVER BEND STATION
STATION OPERATING MANUAL
***SURVEILLANCE TEST PROCEDURE**

*POWER DISTRIBUTION SYSTEM OPERABILITY CHECK

PROCEDURE NUMBER:	***STP-302-0102**
REVISION NUMBER:	***017**
Effective Date:	*** 6/23/2009**

NOTE : SIGNATURES ARE ON FILE.

*INDEXING INFORMATION

TABLE OF CHANGES

LETTER DESIGNATION TRACKING NUMBER	DETAILED DESCRIPTION OF CHANGE(S)

TABLE OF CONTENTS

1 PURPOSE/APPLICABILITY

1.1 Purpose

1.1.1. To verify correct breaker alignments and voltage to required AC, DC and AC Vital bus electrical power distribution subsystems. This satisfies Technical Specifications SR 3.8.9.1 or SR 3.8.10.1.

1.1.2. To verify correct inverter voltage, frequency, and alignments to required AC Vital Buses. This satisfies Technical Specification SR 3.8.7.1 or SR 3.8.8.1

1.2 Applicability

1.2.1. Modes 1, 2, and 3 (TS 3.8.7 & TS 3.8.9)

1.2.2. Modes 4, 5
During movement of recently irradiated fuel assemblies in the primary containment or fuel building (TS 3.8.8 & TS 3.8.10)

2 <u>REFERENCES</u>

2.1 RBS USAR Section 8.3.1 and USAR Section 8.3.2

2.2 RBS Technical Specifications Section 3.8.7, 3.8.8, 3.8.9 and 3.8.10, Distributions Systems

2.3 Drawings:

2.3.1. EE-1M, Standby Bus E22-S004

2.3.2. EE-1ZC, Standby Bus A & B Low Voltage Distribution System

2.3.3. EE-1ZG, Standby Bus A ENB-SWG01A, ENB-PNL02A, 03A

2.3.4. EE-1ZH, Standby Bus B ENB-SWG01B, ENB-PNL02B, 03B

2.3.5. EE-1ZJ, Normal Standby Backup Charger Sys

3 DEFINITIONS

None

4 REQUIRED EQUIPMENT

4.1 Multimeter, if required for 7.1.3 or 7.2.3.

5 PRECAUTIONS AND LIMITATIONS

5.1 The OSM/CRS should be immediately notified and Technical Specification LCO 3.8.7, 3.8.8, 3.8.9 or 3.8.10 referred to if any acceptance criteria can not be met.

5.2 The NCO should be immediately notified if a breaker is found out of position. Any breaker found out of position shall not be repositioned without permission from the OSM/CRS.

5.3 The ENB Battery Chargers must be in the float mode to ensure the proper ENB switchgear bus voltage readings are obtained.

5.4 This procedure has been separated by divisions to allow for a single division to be performed.

5.5 Voltage readings found to be out of the band given in the procedure require the voltage and the bus to be recorded under the comments section on the SURVEILLANCE TEST COVER SHEET.

5.6 All Steps within a Section must be completed to satisfy Technical Specification requirements for the respective division.

5.7 Unless otherwise indicated, steps within this procedure may be performed in any order.

6 PREREQUISITES

6.1 Check that personnel performing this test meet the qualification of ADM-0007, Selection, Training, Qualification and Evaluation of Plant Staff Personnel.

______ *(Initials)*

6.2 Each performer indicates that he has read and understands this procedure by completing the following:

______ (Signature)	______ (Print Name)	______ (Initials)
______ (Signature)	______ (Print Name)	______ (Initials)
______ (Signature)	______ (Print Name)	______ (Initials)
______ (Signature)	______ (Print Name)	______ (Initials)
______ (Signature)	______ (Print Name)	______ (Initials)
______ (Signature)	______ (Print Name)	______ (Initials)

6.3 Verify this procedure is the latest revision.

______ *(Initials)*

6.4 Obtain OSM/CRS permission to begin this test.

______ *(Initials)*

6.5 Inform NCO of test performance.

______ *(Initials)*

NOTE

All readings and checks may be completed in any order

7 PROCEDURE

7.1 Division I Power Distribution Verification

7.1.1. At H13-P877, check the following:

1. ENS-SWG1A, 4.16 KV STBY BUS voltage is 3960 VAC to 4360 VAC as indicated on meter V-1EGSA08, STBY BUS A VOLTS.

   _________Volts ____________ *(Initials)*

2. One of the following breakers is CLOSED for ENS-SWG1A, 4.16 KV STBY BUS:

   1. ENS-ACB06, NORMAL SUPPLY BRKR ____________ *(Initials)*

   2. ENS-ACB04, ALTERNATE SUPPLY BRKR ____________ *(Initials)*

3. EJS-ACB17, EJS-SWGR 1A SPLY BRKR is CLOSED. ____________ *(Initials)*

4. EJS-ACB38, EJS-SWGR 2A SPLY BRKR is CLOSED. ____________ *(Initials)*

7.1.2. At H13-P808, check the following:

1. ENB-SWG01A, 125 VDC STBY BUS A voltage is 130 VDC to 140 VDC as indicated on meter ENB-SWG01A-V, BUS VOLT METER.

   _________Volts ____________ *(Initials)*

2. ENB-ACB560, PRIMARY CHGR BRKR is CLOSED. ____________ *(Initials)*

3. ENB-ACB561, BAT 1A SPLY BRKR is CLOSED. ____________ *(Initials)*

NOTE

Perform either step 7.1.3.1 OR steps 7.1.3.2 - 7.1.3.6 to check the voltage at VBS-PNL01A.

7.1.3. At VBS-PNL01A in the Main Control Room, perform the following:

1. At VBS-PNL01A voltage is 116.6 VAC to 122.4 VAC as indicated on meter VBS-PNL01A V1.

   _________VAC

   _____________ *(Initials)*

2. Close Spare circuit 16 disconnect.

   _____________ *(Initials)*

3. I&C/EM measure voltage on Spare circuit 16 disconnect.

   _________VAC

   _____________ *(Initials)*

4. Record test equipment used in Step 7.1.3.3.

   _____________ *(Initials)*

M&TE ID Number	Calibration Due Date

5. Check VBS-PNL01A voltage is 116.6 VAC to 122.4 VAC as measured at Spare circuit 16 disconnect.

   _____________ *(Initials)*

6. Open spare circuit 16 disconnect.

   _____________ *(Initials)*

_______________ ***(IND VERIF)***

7.1.4. In Stby Swg Rm 1A 98' el Control Bldg, check the following:

1. At ENS-SWG1A, 4.16 KV STBY BUS 1A:

   1. ACB05, EJS-X1A LOAD CENTER XFMR BRKR, is CLOSED.

   ___________ *(Initials)*

   2. ACB10, EJS-X2A LOAD CENTER XFMR BRKR, is CLOSED.

   ___________ *(Initials)*

   3. ACB01, EJS-X3A LOAD CENTER XFMR BRKR, is CLOSED.

   ___________ *(Initials)*

2. At EJS-SWG1A, 480 V BUS:

   1. Bus voltage is 450 VAC to 510 VAC as indicated on EJS-SWG1A VM.

   ___________ *(Initials)*

   _________Volts

   2. BKR EJS-ACB017, CKT. #1EJSA01 STANDBY BUS 1A SUPPLY BKR, is CLOSED.

   ___________ *(Initials)*

   3. BKR EJS-ACB08, EHS-MCC14A, STANDBY SWGR ROOM 1A, is CLOSED.

   ___________ *(Initials)*

   4. BKR EJS-ACB010, ENB-CHGR1A CONTROL BLDG. STANDBY BATTERY CHARGER, is CLOSED.

   ___________ *(Initials)*

   5. BKR EJS-ACB014, EHS-MCC8A STANDBY SWGR ROOM 1A, is CLOSED.

   ___________ *(Initials)*

3. At EHS-MCC8A, 480 VOLT STANDBY MCC:

    1. One of the following breakers is ON for the operating inverter:

    - BRKR 3AB, ENB-INV01A 480 VAC NORMAL ENB-INV01A VITAL BUS A INVERTER

      ____________ *(Initials)*

    OR

    - BRKR 7AB, ENB-INV01A1 VITAL BUS A INVERTER

      ____________ *(Initials)*

    2. BRKR 2AB, SCV-XD8A1 STANDBY DISTRIBUTION TRANSFORMER is ON.

    ____________ *(Initials)*

4. MAIN BREAKER inside SCV-PNL8A1 PANEL is ON.

    ____________ *(Initials)*

5. MAIN BREAKER inside SCV-PNL14A1 PANEL is ON.

    ____________ *(Initials)*

6. At EHS-MCC14A, 480 VOLT STANDBY MCC:

    1. BRKR 1BT, SCV-XD14A1 STANDBY SWGR STANDBY DIST TRANSFORMER is ON.

    ____________ *(Initials)*

    2. One of the following breakers is ON for the operating inverter:

    - For ENB-INV01A, BRKR 2AT, ENB-INV01A 480 VAC BYP ENB-INV01A VITAL BUS A INVERTER

      ____________ *(Initials)*

    OR

    - For ENB-INV01A1, BRKR 2DB, ENB-INV01A1 VITAL BUS A INVERTER

      ____________ *(Initials)*

3. BRKR 2B, SCM-XRC14A1 STANDBY TRANSFORMER FOR SCM-XD14A11 is ON.

_____________ *(Initials)*

7. At ENB-SWG01A, 125 VDC STANDBY SWITCHGEAR:

1. One of the following breakers is closed for the operating inverter:

- For ENB-INV01A, ACB-564, ENB-INV01A VITAL BUS INVERTER

_____________ *(Initials)*

<u>OR</u>

- For ENB-INV01A1, ACB 570, ENB-INV01A1 VITAL BUS INVERTER

_____________ *(Initials)*

2. ACB-567, ENB-PNL02A CONTROL ROOM is CLOSED.

_____________ *(Initials)*

3. ACB-565, ENB-MCC1 125 VDC is CLOSED.

_____________ *(Initials)*

4. ACB-569, ENB-PNL04A CONTROL BLDG is CLOSED.

_____________ *(Initials)*

5. ACB-568, ENB-PNL03A DIESEL BLDG is CLOSED.

_____________ *(Initials)*

6. ACB-560, ENB-CHGR1A BAT CHARGER 300A CONTROL BLDG is CLOSED.

_____________ *(Initials)*

NOTE

Either ENB-INV01A (Section 7.1.5) OR ENB-INV01A1 (Section 7.1.6) shall be operable.

7.1.5. IF ENB-INV01A is in service, THEN at ENB-INV01A Inverter, 116' el. Control Bldg, check the following:

1. ENB-INV01A, the INVERTER SUPPLYING LOAD light is lit, indication the Static switch is in the inverter position.

   ____________ *(Initials)*

2. ENB-INV01A, MANUAL BYPASS SWITCH is in the NORMAL OPERATION position.

   ____________ *(Initials)*

3. Output Voltage is 118 VAC to 123 VAC as indicated on ENB-INV01A, SYSTEM OUTPUT.

   ____________ *(Initials)*

   __________Volts

4. Output frequency is 59.5 Hz to 60.5 Hz as indicated on ENB-INV01A, SYSTEM OUTPUT.

   ____________ *(Initials)*

   __________Hz

5. AC INPUT TO RECTIFIER is ON.

   ____________ *(Initials)*

6. BATTERY INPUT is ON.

   ____________ *(Initials)*

7. INVERTER OUTPUT is ON.

   ____________ *(Initials)*

8. SYSTEM OUTPUT is ON.

   ____________ *(Initials)*

9. BYPASS SOURCE AC INPUT is ON.

   ____________ *(Initials)*

10. At VBS-TRS02A, MANUAL TRANSFER SWITCH on 98' el. Control Bldg, check the following:

    1. The MANUAL TRANSFER SWITCH is in the ENB-INV01A TO LOAD position.

    ____________ *(Initials)*

    2. The ALARM SELECTOR SWITCH is in the COMMON ALARM FROM ENB-INV01A position.

    ____________ *(Initials)*

7.1.6. <u>IF</u> ENB-INV01A1 is in service, <u>THEN</u> at ENB-INV01A1 inverter, 98' el. Control Bldg, check the following:

1. ENB-INV01A1, the INVERTER SUPPLYING LOAD light is lit, indication the Static switch is in the inverter position.

   ____________ *(Initials)*

2. ENB-INV01A1, MANUAL BYPASS SWITCH is in the NORMAL OPERATION position.

   ____________ *(Initials)*

3. Output Voltage is 118 VAC to 123 VAC as indicated on ENB-INV01A1, SYSTEM OUTPUT.

   ____________ *(Initials)*

   _________Volts

4. Output frequency is 59.5 Hz to 60.5 Hz as indicated on ENB-INV01A1, SYSTEM OUTPUT.

   ____________ *(Initials)*

   _________Hz

5. AC INPUT TO RECTIFIER is ON.

   ____________ *(Initials)*

6. BATTERY INPUT is ON.

   ____________ *(Initials)*

7. INVERTER OUTPUT is ON.

   ____________ *(Initials)*

8. SYSTEM OUTPUT is ON.

(Initials)

9. BYPASS SOURCE AC INPUT is ON.

(Initials)

10. At VBS-TRS02A, MANUAL TRANSFER SWITCH on 98' el. Control Bldg, check the following:

    1. The MANUAL TRANSFER SWITCH is in the ENB-INV01A1 TO LOAD position.

    (Initials)

    2. The ALARM SELECTOR SWITCH is in the COMMON ALARM FROM ENB-INV01A1 position.

    (Initials)

7.1.7. At ENB-CHRG1A DIV I 125 VDC STANDBY BAT CHGR, 116' el. Control Bldg, check the following:

1. AC POWER BRKR is ON.

(Initials)

2. DC POWER BRKR is ON.

(Initials)

7.1.8. At Diesel Generator Room A, check the following:

1. At EHS-MCC15A, 480 VOLT STANDBY MCC, BRKR 1BB, SCV-XD15A1 STBY DISTRIBUTION TRANSFORMER is ON.

(Initials)

2. MAIN BREAKER inside SCV-PNL15A1 PANEL is ON.

(Initials)

7.1.9. In the Standby Service Water Pumphouse, check the following:

1. In Pumproom A 118' el. at EHS-MCC16A, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD16A1 STANDBY COOLING TOWER STBY DISTRIBUTION XFMR is ON.

_____________ *(Initials)*

2. At 137' el, the MAIN BREAKER inside SCV-PNL16A1 PANEL is ON.

_____________ *(Initials)*

7.1.10. In the Auxiliary Bldg 114' el (west), check the following:

1. At EHS-MCC2E, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD2E1 AUXILIARY BUILDING STBY DIST XFMR is ON.

_____________ *(Initials)*

2. At EHS-MCC2G, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD2G1 AUXILIARY BUILDING STBY DIST XFMR is ON.

_____________ *(Initials)*

3. MAIN BREAKER inside SCV-PNL2E1 PANEL is ON. (south of EHS-MCC2G)

_____________ *(Initials)*

4. MAIN BREAKER inside SCV-PNL2G1 PANEL is ON. (west of EHS-MCC2E)

_____________ *(Initials)*

7.1.11. In the Auxiliary Bldg, 141' el (west), check the following:

1. At EJS-SWG2A, 480 V STANDBY SWITCHGEAR:

1. Bus voltage is 450 VAC to 510 VAC as indicated on EJS-SWG2A VM.

_____________ *(Initials)*

_________Volts

2. BKR EJS-ACB027, EHS-MCC2A AUX BLDG is CLOSED.

_____________ *(Initials)*

3. BKR EJS-ACB028, EHS-MCC2C AUX BLDG is CLOSED.

_______________ *(Initials)*

4. BKR EJS-ACB029, EHS-MCC2E AUX BLDG is CLOSED.

_______________ *(Initials)*

5. BKR EJS-ACB030, EHS-MCC2G AUX BLDG is CLOSED.

_______________ *(Initials)*

6. BKR EJS-ACB031, EHS-MCC2J AUX BLDG is CLOSED.

_______________ *(Initials)*

7. BKR EJS-ACB035, EHS-MCC2L AUX BLDG is CLOSED.

_______________ *(Initials)*

8. BKR EJS-ACB034, EHS-MCC15A DIESEL GEN RM A is CLOSED.

_______________ *(Initials)*

2. At EHS-MCC2A, 480 VOLT STANDBY MCC, BRKR 1D, SCV-XD2A1 AUXILIARY BUILDING STBY DISTRIBUTION XFMR is ON.

_______________ *(Initials)*

3. At EHS-MCC2A, 480 VOLT STANDBY MCC, BRKR 1CB, SCV-XD2A2 AUXILIARY BLDG STBY DIST XFMR is ON.

_______________ *(Initials)*

4. At EHS-MCC2C, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD2C1 AUXILIARY BUILDING DISTRIBUTION XFMR is ON.

_______________ *(Initials)*

5. At EHS-MCC2J, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD2J1 AUXILIARY BLDG STBY DISTRIBUTION XFMR is ON.

_______________ *(Initials)*

6. At EHS-MCC2L, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD2L1 AUXILIARY BUILDING STBY DISTRIBUTION XFMR is ON.

   __________ *(Initials)*

7. MAIN BREAKER inside SCV-PNL2A1 PANEL is ON. (north wall, north of EHS-MCC2A)

   __________ *(Initials)*

8. MAIN BREAKER inside SCV-PNL2A2 PANEL is ON. (south of EHS-MCC2C)

   __________ *(Initials)*

9. MAIN BREAKER inside SCV-PNL2C1 PANEL is ON. (25 feet east of SCV-PNL2A2)

   __________ *(Initials)*

10. MAIN BREAKER inside SCV-PNL2J1 PANEL is ON. (behind EJS-SWG2A)

   __________ *(Initials)*

11. MAIN BREAKER inside SCV-PNL2L1 PANEL is ON. (behind EJS-SWG2A)

   __________ *(Initials)*

7.2 Division II Power Distribution Verification

7.2.1. At H13-P877, check the following:

1. ENS-SWG1B, 4.16 KV STBY BUS has 3960 VAC to 4360 VAC as indicated on meter V-EGSB08, STBY BUS B VOLTS.

_________Volts ____________ *(Initials)*

2. One of the following breakers is CLOSED for ENS-SWG1B, 4.16 KV BUS:

1. ENS-ACB26, NORMAL SUPPLY BRKR ____________ *(Initials)*

2. ENS-ACB24, ALTERNATE SPLY BRKR ____________ *(Initials)*

3. EJS-ACB41, EJS-SWGR 1B SPLY BRKR is CLOSED. ____________ *(Initials)*

4. EJS-ACB78, EJS-SWGR 2B SPLY BRKR is CLOSED. ____________ *(Initials)*

7.2.2. At H13-P808, check the following:

1. ENB-SWG01B, 125 VDC STBY BUS B voltage is 130 VDC to 140 VDC as indicated on meter ENB-SWG01B-V, BUS VOLT METER.

_________Volts ____________ *(Initials)*

2. ENB-ACB580, PRIMARY CHGR BRKR is CLOSED. ____________ *(Initials)*

3. ENB-ACB581, BAT 1B SPLY BRKR is CLOSED. ____________ *(Initials)*

NOTE

Perform either step 7.2.3.1 OR steps 7.2.3.2 - 7.2.3.6 to check the voltage at VBS-PNL01B.

7.2.3. At VBS-PNL01B, in the Main Control Room, perform the following:

1. At VBS-PNL01B, voltage is 116.6 to 122.4 VAC as indicated on meter VBS-PNL01B V1.

   _________VAC

   ____________ *(Initials)*

2. Close Spare circuit 25 disconnect.

   ____________ *(Initials)*

3. I&C/EM measure voltage on Spare circuit 25 disconnect.

   _________VAC

   ____________ *(Initials)*

4. Record test equipment used in Step 7.2.3.3.

   ____________ *(Initials)*

M&TE ID Number	Calibration Due Date

5. Check VBS-PNL01B voltage is 116.6 VAC to 122.4 VAC as measured at Spare circuit 25 disconnect.

   ____________ *(Initials)*

6. Open spare circuit 25 disconnect.

   ____________ *(Initials)*

____________ ***(IND VERIF)***

7.2.4. In Stby Swg Rm 1B, el 98' Control Bldg, check the following:

1. At ENS-SWG1B, 4.16 KV STANDBY BUS B:

   1. ACB25, EJS-X1B LOAD CENTER XFMR BREAKER is CLOSED.

   ________ *(Initials)*

   2. ACB30, EJS-X2B LOAD CENTER XFMR BREAKER is CLOSED.

   ________ *(Initials)*

   3. ACB21, EJS-X3B STBY CLG TWR NO. 1 BREAKER is CLOSED.

   ________ *(Initials)*

2. At EJS-SWG1B, 480 V BUS:

   1. Bus voltage is 450 VAC to 510 VAC as indicated on EJS-SWG1B VM.

   _________Volts

   ________ *(Initials)*

   2. BKR EJS-ACB041, CKT. #1EJSB01 STANDBY BUS 1B SUPPLY BREAKER is CLOSED.

   ________ *(Initials)*

   3. BRKR EJS-ACB051, ENB-CHGR1B CONTROL BLDG STANDBY BATTERY CHARGER is CLOSED.

   ________ *(Initials)*

   4. BRKR EJS-ACB055, EHS-MCC8B STANDBY SWGR ROOM 1B is CLOSED.

   ________ *(Initials)*

   5. BKR EJS-ACB048, EHS-MCC14B STANDBY SWGR ROOM 1B is CLOSED.

   ________ *(Initials)*

3. At EHS-MCC8B, 480 VOLT STANDBY MCC:

   1. One of the following breakers is ON for the operating inverter:

      - For ENB-INV01B, BRKR 3AB, ENB-INV01B 480 VAC NORMAL ENB-INV01B VITAL BUS B INVERTER

        _______ *(Initials)*

      OR

      - For ENB-INV01B1, BRKR 6AB ENB-INV01B1 VITAL BUS B INVERTER

        _______ *(Initials)*

   2. BRKR 2AB, SCV-XD8B1 STANDBY DISTRIBUTION TRANSFORMER is ON.

      _______ *(Initials)*

4. MAIN BREAKER inside SCV-PNL8B1 PANEL is ON.

   _______ *(Initials)*

5. MAIN BREAKER inside SCV-PNL14B1 PANEL is ON.

   _______ *(Initials)*

6. At EHS-MCC14B, 480 VOLT STANDBY MCC:

   1. BRKR 3AT, SCV-XD14B1 STANDBY SWGR STANDBY DIST TRANSFORMER is ON.

      _______ *(Initials)*

   2. One of the following breakers is ON for the operating inverter:

      - For ENB-INV01B, BRKR 4AT, ENB-INV01B 480 VAC BYP ENB-INV01B VITAL BUS B INVERTER

        _______ *(Initials)*

      OR

      - For ENB-INV01B1, BRKR 3BB, ENB-INV01B1 VITAL BUS B INVERTER

        _______ *(Initials)*

3. BRKR 3C, SCM-XRC14B1 STANDBY TRANSFORMER is ON.

_______ *(Initials)*

7. At ENB-SWG01B, 125 VDC STANDBY SWITCHGEAR:

1. One of the following breakers is closed for the operating inverter:

- For ENB-INV01B, ACB584, ENB-INV01B VITAL BUS INVERTER BRKR is CLOSED.

_______ *(Initials)*

<u>OR</u>

- For ENB-INV01B1, ACB588, ENB-INV01B1 VITAL BUS INVERTER BRKR is CLOSED.

_______ *(Initials)*

2. ACB586, ENB-PNL02B CONTROL ROOM BRKR is CLOSED.

_______ *(Initials)*

3. ACB587, ENB-PNL03B DIESEL BLDG BRKR is CLOSED.

_______ *(Initials)*

4. ACB580, ENB-CHRG1B BAT. CHARGER 300A CONTROL BLDG BRKR is CLOSED.

_______ *(Initials)*

NOTE

Either ENB-INV01B (Section 7.2.5) OR ENB-INV01B1 (Section 7.2.6) shall be operable.

7.2.5. IF ENB-INV01B is in service, THEN at ENB-INV01B UPS Inverter, 116' el Control Bldg, check the following:

1. ENB-INV01B, the INVERTER SUPPLYING LOAD light is lit, indication the Static switch is in the inverter position.

   _______ *(Initials)*

2. ENB-INV01B, MANUAL BYPASS SWITCH is in the NORMAL OPERATION position.

   _______ *(Initials)*

3. Output Voltage is 118 VAC to 123 VAC as indicated on ENB-INV01B, SYSTEM OUTPUT.

   _______ *(Initials)*

   _________Volts

4. Output frequency is 59.5 Hz to 60.5 Hz as indicated on ENB-INV01B, SYSTEM OUTPUT.

   _______ *(Initials)*

   _________Hz

5. AC INPUT TO RECTIFIER is ON.

   _______ *(Initials)*

6. BATTERY INPUT is ON.

   _______ *(Initials)*

7. INVERTER OUTPUT is ON.

   _______ *(Initials)*

8. SYSTEM OUTPUT is ON.

   _______ *(Initials)*

9. BYPASS SOURCE AC INPUT is ON.

   _______ *(Initials)*

10. At VBS-TRS02B, MANUAL TRANSFER SWITCH on 98' el Control Bldg, check the following:

    1. The MANUAL TRANSFER SWITCH is in the ENB-INV01B TO LOAD POSITION.

    ___________ *(Initials)*

    2. The ALARM SELECTOR SWITCH is in the COMMON ALARM FROM ENB-INV01B position.

    ___________ *(Initials)*

7.2.6. <u>IF</u> ENB-INV01B1 is in service, <u>THEN</u> at ENB-INV01B1 inverter, 98’ el Control Bldg, check the following:

1. ENB-INV01B1, the INVERTER SUPPLYING LOAD light is lit, indication the Static switch is in the inverter position.

   ___________ *(Initials)*

2. ENB-INV01B1, MANUAL BYPASS SWITCH is in the NORMAL OPERATION position.

   ___________ *(Initials)*

3. Output Voltage is 118 VAC to 123 VAC as indicated on ENB-INV01B1, SYSTEM OUTPUT.

   ___________ *(Initials)*

   _________Volts

4. Output frequency is 59.5 Hz to 60.5 Hz as indicated on ENB-INV01B1, SYSTEM OUTPUT.

   ___________ *(Initials)*

   _________Hz

5. AC INPUT TO RECTIFIER is ON.

   ___________ *(Initials)*

6. BATTERY INPUT is ON.

   ___________ *(Initials)*

7. INVERTER OUTPUT is ON.

   ___________ *(Initials)*

8. SYSTEM OUTPUT is ON.

   ___________ *(Initials)*

9. BYPASS SOURCE AC INPUT is ON. ____________
(Initials)

10. At VBS-TRS02B, MANUAL TRANSFER SWITCH on 98' el Control Bldg, check the following:

    1. The MANUAL TRANSFER SWITCH is in the ENB-INV01B1 TO LOAD POSITION. ____________
    (Initials)

    2. The ALARM SELECTOR SWITCH is in the COMMON ALARM FROM ENB-INV01B1 position. ____________
    (Initials)

7.2.7. At ENB-CHRG1B, DIV II 125 VDC STANDBY BAT CHGR 116' el Control Bldg, check the following:

1. AC POWER BRKR is ON. ____________
(Initials)

2. DC POWER BRKR is ON. ____________
(Initials)

7.2.8. At Diesel Generator Room B, check the following:

1. At EHS-MCC15B, 480 VOLT STANDBY MCC, BRKR 1BB, SCV-XD15B1 STBY DISTRIBUTION TRANSFORMER is ON. ____________
(Initials)

2. MAIN BREAKER inside SCV-PNL15B1 PANEL is ON. ____________
(Initials)

7.2.9. In the Standby Service Water Pumphouse, check the following:

1. In Pumproom B 118' el, at EHS-MCC16B, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD16B1 STANDBY COOLING TOWER STBY DISTRIBUTION XFMR is ON. ____________
(Initials)

2. At 137' el, the MAIN BREAKER inside SCV-PNL16B1 PANEL is ON. ____________
(Initials)

7.2.10. In the Auxiliary Bldg 114' el (east), check the following:

1. At EHS-MCC2H, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD2H1 AUXILIARY BUILDING STBY DIST XFMR is ON.

 ______ *(Initials)*

2. At EHS-MCC2F, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD2F1 AUXILIARY BUILDING STBY DIST XFMR is ON.

 ______ *(Initials)*

3. MAIN BREAKER inside SCV-PNL2H1 PANEL is ON. (east wall Aux. Bldg. behind EHS-MCC2H)

 ______ *(Initials)*

4. MAIN BREAKER inside SCV-PNL2F1 PANEL is ON. (east wall Aux. Bldg. behind EHS-MCC2H)

 ______ *(Initials)*

7.2.11. In the Auxiliary Bldg 141' el (east), check the following:

1. At EJS-SWG2B, 480 V STANDBY SWITCHGEAR:

   1. Bus voltage is 450 VAC to 510 VAC as indicated on EJS-SWG2B VM.

 ______ *(Initials)*

 ________Volts

   2. BKR EJS-ACB067, EHS-MCC2B AUX BLDG is CLOSED.

 ______ *(Initials)*

   3. BKR EJS-ACB068, EHS-MCC2D AUX BLDG is CLOSED.

 ______ *(Initials)*

   4. BKR EJS-ACB069, EHS-MCC2F AUX BLDG is CLOSED.

 ______ *(Initials)*

   5. BKR EJS-ACB070, EHS-MCC2H AUX BLDG is CLOSED.

 ______ *(Initials)*

6. BKR EJS-ACB071, EHS-MCC2K AUX BLDG is CLOSED.

______ *(Initials)*

7. BKR EJS-ACB074, EHS-MCC15B DIESEL GEN RM B is CLOSED.

______ *(Initials)*

2. At EHS-MCC2B, 480 VOLT STANDBY MCC, BRKR 3A, SCV-XD2B1 AUXILIARY BUILDING STBY DISTRIBUTION XFMR is ON.

   ______ *(Initials)*

3. At EHS-MCC2B, 480 VOLT STANDBY MCC, BRKR 1CT, SCV-XD2B2 AUXILIARY BUILDING STBY DIST XFMR is ON.

   ______ *(Initials)*

4. At EHS-MCC2D, 480 VOLT STANDBY MCC, BRKR 1BT, SCV-XD2D1 AUXILIARY BLDG STBY DISTRIBUTION XFMR is ON.

   ______ *(Initials)*

5. At EHS-MCC2K, 480 VOLT STANDBY MCC, BRKR 7AT, SCV-XD2K1 AUXILIARY BUILDING STBY DISTRIBUTION XFMR is ON.

   ______ *(Initials)*

6. MAIN BREAKER inside SCV-PNL2B1 PANEL is ON. (north of elevator)

   ______ *(Initials)*

7. MAIN BREAKER inside SCV-PNL2B2 PANEL is ON. (east of EJS-SWG2B)

   ______ *(Initials)*

8. MAIN BREAKER inside SCV-PNL2D1 PANEL is ON. (east of EJS-SWG2B)

   ______ *(Initials)*

9. MAIN BREAKER inside SCV-PNL2K1 PANEL is ON. (east wall)

   ______ *(Initials)*

7.3 Division III Power Distribution Verification

7.3.1. At H13-P601, check the following:

1. DIV III E22-S004 4.16 KV BUS voltage is 3960 VAC to 4360 VAC on meter E22-R610, HPCS 4160 V BUS VOLTS.

 ____________ *(Initials)*

 _________Volts

2. E22-ACB04, HPCS BUS SUPPLY BRKR is CLOSED.

 ____________ *(Initials)*

3. E22-ACB03, HPCS MCC SUPPLY BRKR is CLOSED.

 ____________ *(Initials)*

4. DIV III E22-S002 480 V MCC voltage is 452 VAC to 524 VAC on meter E22-R617, HPCS 480 V MCC VOLTS.

 ____________ *(Initials)*

 _________Volts

7.3.2. At H13-P808, check the following:

1. STBY BATT C E22-S001 CHGR voltage is 130 VDC to 140 VDC on meter E22-S001BAT-V, BANK (HPCS) VOLT METER.

 ____________ *(Initials)*

 _________Volts

2. E22-ACB620, PRIMARY CHGR BRKR is CLOSED.

 ____________ *(Initials)*

3. E22-ACB621, S001 BAT SUPPLY BRKR is CLOSED.

 ____________ *(Initials)*

7.3.3. At 116' el Control Bldg, check the following:

1. In the Division III Switchgear Room at E22-S002, ESF DIVISION 3 HPCS MOTOR CONTROL CENTER:

   1. BRKR 3AL, HPCS BATTERY CHARGER NO. 1 E22-S001 CGR is ON.

   ____________ *(Initials)*

   2. BRKR 3AR, DISTRIBUTION TRANSFORMER FEEDER BREAKER is ON.

   ____________ *(Initials)*

2. In the Division III Battery Charger Room, check the following:

   1. Battery Charger AC BRKR is ON.

   ____________ *(Initials)*

   2. Battery Charger DC BRKR is ON.

   ____________ *(Initials)*

   3. E22-SW1, E22-S001BAT BATTERY Fused Disconnect Switch is ON.

   ____________ *(Initials)*

7.3.4. In the Division III Diesel Generator Control Room, check the following:

1. Inside E22-PNLS001, CB 9, BATTERY CHARGER is ON.

   ____________ *(Initials)*

2. At E22-PNLS001, voltage is 130 VDC to 140 VDC as indicated on meter M18.

   ____________ *(Initials)*

_________Volts

7.4 Notify NCO of test completion.

____________ *(Initials)*

7.5 Notify OSM/CRS of test completion.

____________ *(Initials)*

8 ACCEPTANCE CRITERIA

8.1 Modes 1, 2, and 3

8.1.1. Requirements of Technical Specification SR 3.8.9.1, verification of correct breaker alignment and voltage to the required AC, DC, and AC vital bus electrical power distribution subsystems, have been satisfied by completion of the following steps:

☐ YES ☐ NO ☐ N/A

STEP	✓
7.1	
7.2	
7.3	

8.1.2. Requirements of Technical Specification SR 3.8.7.1, verification of correct inverter voltage, frequency, and alignments to required AC vital buses, have been satisfied by completion of the following steps:

☐ YES ☐ NO ☐ N/A

STEP	✓
7.1.5 OR 7.1.6	
7.2.5 OR 7.2.6	

8.2 Modes 4 and 5 and during movement of recently irradiated fuel assemblies in the Primary Containment or Fuel Building,

8.2.1. Requirements of Technical Specification SR 3.8.10.1, verification of correct breaker alignment and voltage to the required AC, DC, and AC vital bus electrical power distribution subsystems, have been satisfied by completion of the following steps:

NOTE

That portion of either Division not required to power required operable equipment may be deenergized and meet acceptance criteria.

1. For Div I when required to be operable.

☐ YES ☐ NO ☐ N/A

STEP	✓
7.1	

2. For Div II when required to be operable.

☐ YES ☐ NO ☐ N/A

STEP	✓
7.2	

3. For Div III when required to be operable.

☐ YES ☐ NO ☐ N/A

STEP	✓
7.3	

8.2.2. Requirements of Technical Specification SR 3.8.8.1, verification of correct inverter voltage, frequency, and alignments to required AC vital buses, have been satisfied by completion of the following steps:

1. When Div I is required to be operable.

☐ YES ☐ NO ☐ N/A

STEP	✓
7.1.5 OR 7.1.6	

2. When Div II is required to be operable.

☐ YES ☐ NO ☐ N/A

STEP	✓
7.2.5 OR 7.2.6	

8.3 The test is acceptable by satisfying the above criteria.

☐ YES ☐ NO

______________________________ Signature/KCN

______________ Date/Time

9 RECORDS

9.1 Upon completion of this test, forward the entire Procedure Data Package to the OSM/CRS for review and approval.

9.2 Upon completion of the entire review process, records disposition shall be in accordance with ADM-0015, Station Surveillance Test Program.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 600 psig.	A.1	----------NOTE---------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. -----------------------------	
		Declare the associated control rod scram time "slow."	8 hours
	<u>OR</u>		
	A.2	Declare the associated control rod inoperable.	8 hours

(continued)

ACTION (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 600 psig.	B.1 Restore charging water header pressure to ≥ 1540 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 1540 psig
	<u>AND</u>	
	B.2.1 -----------NOTE----------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ------------------------------- Declare the associated control rod scram time "slow."	1 hour
	<u>OR</u>	
	B.2.2 Declare the associated control rod inoperable.	1 hour
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 600 psig.	C.1 Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 1540 psig
	<u>AND</u>	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Declare the associated control rod inoperable.	1 hour
D. Required Action and associated Completion Time of Required Action B.1 or C.1 not met.	D.1 ----------NOTE----------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. ----------------------------- Place the reactor mode switch in the shutdown position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is $\geq$ 1540 psig.	7 days