2018 NRC Written Answer Key

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

U.S. Nuclear Regulatory Commission

Site-Specific RO Written Examination

Applicant Information	
Name:	
Date:	Facility/Unit: **River Bend Station**
Region: I ☐ II ☐ III ☐ IV ■	Reactor Type: W ☐ CE ☐ BW ☐ GE ■
Start Time:	Finish Time:

Instructions

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

Applicant Certification

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

____
Applicant's Signature

Results		
Examination Value	___________	Points
Applicant's Score	___________	Points
Applicant's Grade	___________	Percent

Examination Outline Cross Reference	Level	RO
203000 RHR/LPCI: Injection Mode A2. Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.14 Initiating logic failure	Tier	2
	Group #	1
	K/A	203000 A2.14
	Rating	3.8
	Revision	3
Revision Statement: 3: Modified second part.		

Question: 1

The crew is performing an Emergency Depressurization due to low reactor water level resulting from a drywell leak.

Reactor Vessel Water Level Low LIS-B21-N691A and High Drywell Pressure PIS-B21-N694A are INOP.

RHR A initiating logic ___(1)___ generate automatic initiation.

Per OSP-53, Emergency and Transient Response Support Procedure, the crew should verify E12-F042A, RHR PUMP A LPCI INJECT ISOL VALVE, opens when RPV pressure decreases below ___(2)___ psig (MAXIMUM).

	(1)	(2)
A.	can	225
B.	can	487
C.	cannot	225
D.	cannot	487

Answer: D

Explanation:

RHR A Initiation sequence requires two separate signals to auto initiate. There are four separate inputs, Channel A and E High Drywell Pressure (1.68 psid) and Reactor Vessel Water Level Low (Level 1). If channel A High Drywell Pressure and Low Reactor Water level signals are INOP, the initiating logic will not generate an auto start signal.

Per OSP-53, Attachment 7, IF RPV pressure is below 487 psig, THEN perform the following: Verify E12-F042A, RHR PUMP A LPCI INJECT ISOL VALVE, Opens. Per EN-OP-120, Operator Fundamentals Program, operators should verify and report automatic system actuations or response, which includes operator actions if the plant has not responded as expected.

Distracters:

(1) If applicant confuses RHR A logic with HPCS initiating logic, the instrument failures will not prevent an automatic initiation. Per R-STM-203, HPCS, Figure 4, HPCS Initiation Logic, there are separate signals generated based on 4 different RPV Water Level instruments and 4 different Drywell Pressure instruments. One failure in each would not prevent the initiation signal.

(2) 225 psig is plausible because it is the design pressure flow is required to commence. Per R-STM-204, RHR System, Injection into the reactor vessel begins after the injection valves open when reactor pressure decreases to the pump's shutoff head. Design specifications require injection flow to commence at a vessel pressure of 225 psig above drywell pressure.

K/A Match

Applicant must predict the impact of 2 instruments INOP for the RHR A initiation logic. During the emergency depressurization, the applicant must have knowledge of OSP-53 requirement to correctly control RHR A injection valve operation to mitigate the drywell leak.

Technical References:

STM-204, RHR System
R-STM-203, HPCS, Figure 4, HPCS Initiation Logic
OSP-53, Emergency and Transient Response Support Procedure

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0204, Objective 4: List the signals, including setpoints and logic

requirements, that will automatically initiate the Residual Heat Removal System (4)		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(7)	
Level of Difficulty:	4	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Operator Actions: Manual depressurization of reactor vessel		

Examination Outline Cross Reference	**Level**	RO
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) A1. Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including: A1.08 Heat exchanger temperatures	**Tier**	2
	Group #	1
	K/A	205000 A1.08
	Rating	3.1
	Revision	2
Revision Statement: 2: Revised explanation for D.		

Question: 2

The plant is in MODE 4 with the following conditions:

- RHR “B” operating in Shutdown Cooling mode of operation
- Reactor Water Cleanup system (RWCU) is in service
- E12-F048B, RHR B Heat Exchanger Bypass valve is CLOSED
- E12-F003B, RHR B Heat Exchanger Outlet valve is OPEN
- STP-050-0700, RCS Pressure/Temperature Limits Verification has been initiated

Which of the following provides an accurate indication of Reactor Coolant temperature?

A. RWCU Non-Regen heat exchanger inlet temperature G33-RTD-N006

B. RHR B heat exchanger (E12-EB001B) inlet temperature E12-T/C-N004B

C. RPV Pressure, STP-050-0700

D. RHR B heat exchanger discharge temperature E12-T/C-N027B

Answer: B
Explanation: E12-T/C-N004B provides accurate indication of reactor coolant temperature since the heat exchanger outlet valve is open (E12-F003B). The most accurate indication of Reactor Coolant temperature is before the water is cooled in the heat exchanger.

Distracters:

A. Not Correct. With RWCU in service G33-RTD-N006 does not provide an accurate indication of Reactor coolant temperature since this water has already been cooled by RWCU return flow through the Regenerative heat exchanger. This answer is plausible if the candidate is unable to accurately understand the flowpath of the RWCU system

C. Not Correct. Mode 5 pressure indication does not provide an accurate indication of reactor coolant temperature.

D. Not correct. With no reactor coolant flow bypassing the RHR heat exchanger and the heat exchanger outlet valve open (E12-F003B), the discharge temperature indication will be cooled by the heat exchanger. The cooled water will not be an accurate indication of Reactor Coolant temperature. This answer would be correct if E12-F048B was open and E12-F003B was closed.

K/A Match

Applicant must be able to analyze the plant lineup and determine the RHR heat exchanger temperature is the best indication of reactor coolant temperature.

Technical References:

SOP-0031, Residual Heat Removal System,
PID-27-07B, RHR B Piping Diagram, PID-26-03A, RWCU Piping Diagram
STP-050-0700, RCS Pressure/Temperature Limits Verification, Rev 307

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0204, Rev 2, HLO Objective 10: Given key plant/system status and key parameters, predict/determine Residual Heat Removal System response (10)

Question Source:	**Bank #**	
	Modified Bank # June 2016 AUDIT	X

	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	4	
SRO Only Justification: N/A		
PRA Applicability:		

PARENT Question
The plant is in MODE 4 with the following conditions:

- RHR “B” operating in Shutdown Cooling mode of operation
- Reactor Water Cleatem (RWCU) is in service
- E12-F048B, RHR B Heat Exchanger Bypass valve is OPEN
- E12-F003B, RHR B Heat Exchanger Outlet valve is CLOSED
- STP-050-0700, RCS Pressure/Temperature Limits Verification has been initiated

Which of the following provides an accurate indication of Reactor Coolant temperature?

A. RWCU Non-Regen heat exchanger inlet temperature G33-RTD-N006

B. RHR B heat exchanger (E12-EB001B) inlet temperature E12-T/C-N004B

C. RPV Pressure, STP-050-0700

D. RHR B heat exchanger discharge temperature E12-T/C-N027B (CORRECT ANSWER)

Examination Outline Cross Reference	**Level**	RO
209001 Low Pressure Core Spray System K4. Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: K4.09 Load sequencing	**Tier**	2
	Group #	1
	K/A	209001 K4.09
	Rating	3.3
	Revision	1
Revision Statement:		

Question: 3

A simultaneous LOCA and loss of offsite power have occurred.

Per the timed sequenced load guide, which of the following will sequence on first?

A. LPCS PUMP

B. RHR A PUMP

C. RHR B PUMP

D. STDBY GAS TREAT FAN (GTS-FN1A)

Answer: A

Explanation:

Per AOP-4, Loss of Offsite Power, Attachment 1, Division 1 and 2 Timed Sequenced Load Guide, the LPCS pump will be sequenced on in 2 seconds.

Distracters:

Per AOP-4, Loss of Offsite Power, Attachment 1, Division 1 and 2 Timed Sequenced Load Guide, the RHR A and B PUMPS will sequence on in 7 seconds. Both of these are plausible if applicant confuses either of these pumps with RHR C. RHR C is sequenced on in 2 seconds, the same as LPCS PUMP.

Per AOP-4, Loss of Offsite Power, Attachment 1, Division 1 and 2 Timed Sequenced Load Guide, the STDBY GAS TREAT FAN will start in 30 seconds. This is plausible if applicant confuses time sequence with LPCS time sequence.

K/A Match Applicants must have knowledge of load sequence design feature for LPCS system.		
Technical References: Per AOP-4, Loss of Offsite Power, Attachment 1, Division 1 and 2 Timed Sequenced Load Guide		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0205 Objective 19: Given a set of plant conditions, predict the effect that a loss or malfunction of the following will have on the Low Pressure Core Spray System: (19) A.C. Power		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Internal Events: LOSP		

Examination Outline Cross Reference	Level	RO
209001 Low Pressure Core Spray System K6. Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM : K6.01 A.C. power	Tier	2
	Group #	1
	K/A	209001 K6.01
	Rating	3.4
	Revision	1
Revision Statement:		

Question: 4

A loss of power supply 1EHS*MCC2J results in the loss of power to ______.

A. E21-MOVF005, LPCS PUMP INJECTION VALVE

B. E21-PC001, LPCS PUMP

C. E12-MOVF042B, B LPCI INJECTION VALVE

D. E12-C002B, B RHR PUMP

Answer: A

Explanation:

Per R-STM-0205, LPCS, E21-MOVF005 is powered from EHS-MCC2J BKR 3D.

Distracters:

B. Per R-STM-0205, LPCS PUMP is powered from ENS-SWG1A, but is plausible if applicant confuses EHS and ENS power supply.

C . Per R-STM-0204, B LPCI INJECTION VALVE is powered from EHS-MCC2K, but is plausible if applicant confuses MCC2J with MCC2K.

D. Per R-STM-0204, RHR, B RHR PUMP is powered from ENS-SWG1B, but is plausible if applicant confuses EHS and ENS power supplies and division 1 and 2 power supplies.

K/A Match

Applicant must have knowledge to determine the loss of the AC bus was the power supply to the LPCS Injection Valve.		
Technical References: R-STM-0205, LPCS R-STM-0204, RHR		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0205-LO, LPCS, Objective 19: Given a set of plant conditions, predict the effect that a loss or malfunction of the following will have on the Low Pressure Core Spray System: (19) A.C. Power		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(8)	
Level of Difficulty:	2	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
209002 High Pressure Core Spray System (HPCS) K1. 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	Tier	2
	Group #	1
	K/A	209002 K1.04
	Rating	3.8
	Revision	3
Revision Statement: Rev 3: Re-worded distractor A, B, and D.		

Question: 5

The High Pressure Core Spray (HPCS) Diesel Generator is running and paralleled to offsite power to support a surveillance.

The following events occur concurrently:

- A loss of offsite power (LOP)
- A Loss of Coolant Accident (LOCA) signal is generated.

What is the expected response of the HPCS Pump?

A. Automatically starts immediately

B. Automatically starts on a 30-second delay

C. Must be manually started at P601

D. Automatically starts on a 2-second delay

Answer: A
Explanation: For a loss of power condition while the diesel is paralleled to Off-Site Power, the diesel response will vary depending on the initial conditions and how the loss of power occurs. The conditions in the stem indicate that the LOP causes isolation of the safety busses from the electrical distribution system. For this condition, the HPCS D/G will assume the loads of its safety bus without interruption. The LOCA signal will cause pump to start immediately because the diesel generator is up to speed and voltage.

Distracters:

B. The 30 second time delay is plausible because if the applicant confuses electrical sequence with load sequencing; however, the 30 second delay only applies to SSW "C" pump (SWP-P2C) which supplies cooling to the Div 3 DG.

C. This is plausible if the applicant confuses the electrical sequencing with a bus that is lost and must be manually aligned due to previous plant lineup.

D. The 2 second time delay is plausible because if the applicant confuses electrical sequence with load sequencing; however, the 2 second delay only applies to the sequence start for LPCS.

K/A Match

Applicant must have knowledge of HPCS diesel response after a LOP/LOCA while it was already paralleled to the bus. Then the applicant must apply that knowledge and LOP/LOCA conditions to the effect of the HPCS system..

Technical References:

R-STM-0309H Rev 12, Pg 4, 48, 49 of 78
Condition Report: CR-RBS-2006-03776, Discovered a condition where the EDGs may not shift to Emergency mode during a LOP

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0203 Objective 18: Given a set of plant conditions, predict the effect that a loss or malfunction of the following will have on the High Pressure Core Spray (HPCS) System: (18)

Question Source:	**Bank # March 2014 NRC**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	

	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(8)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Internal Events: LOSP		

Examination Outline Cross Reference	Level	RO
209002 High Pressure Core Spray System (HPCS) K5. Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): K5.04 Adequate core cooling: BWR-5,6	**Tier**	2
	Group #	1
	K/A	209002 K5.04
	Rating	3.8
	Revision	2
Revision Statement: 2: Added reference to stem and replaced leak with rupture.		

Question: 6

A LOCA occurred due to a failure of the Reactor Recirculation suction piping.

The rupture has not been isolated.

High Pressure Core Spray is injecting into the core at a rate of 5200 gpm.

What is the lowest reactor water level where adequate core cooling is still maintained per EOP-1, RPV Control?

A. -162 inches

B. -187 inches

C. -200 inches

D. -211 inches

Answer: D
Explanation: Per EOP-1, RPV Control, adequate core cooling is met with Spray Cooling with HPCS or LPCS flow above 5,000 gpm AND RPV level at or above -211 inches.
Distracters: Adequate core Cooling is met with Core Submergence with RPV level above -162 inches. Adequate core Cooling is met with Steam Cooling with Injection with RPV level above -187 inches.

Adequate core Cooling is met with Steam Cooling without Injection with RPV level above -200 inches.
K/A Match Applicant must have knowledge of HPCS requirements to ensure Adequate Core Cooling is met.
Technical References: EOP-0001, RPV Control
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: RLP-HLO-0203-LO HPCS, Objective 17: Given a set of plant conditions, predict the effect that a loss or malfunction of the High Pressure Core Spray (HPCS) System will have on the following: (17) Reactor Water Level Adequate Core Cooling

Question Source:	**Bank # Oct 2012 Audit**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(10)	
Level of Difficulty:	3	

SRO Only Justification: N/A

PRA Applicability:

Examination Outline Cross Reference	Level	RO
211000 Standby Liquid Control System K4. Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: K4.04 Indication of fault in explosive valve firing circuits	Tier	2
	Group #	1
	K/A	211000 K4.04
	Rating	3.8
	Revision	2
Revision Statement: 2: Distractor A added SLC Pump B will start.		

Question: 7

The key shown was turned to RUN position for 2 seconds and returned to NEUTRAL position.

The following indications were observed.

What is the status of SLC?

A. Once the C41-F001B, SLC PUMP B SUCT VLV, opens, then the SQUIB Valve will fire and SLC Pump B will start to commence injecting SLC into the RPV.

B. Once the C41-F001B, SLC PUMP B SUCT VLV, opens, then the SLC Pump B will start to commence injecting SLC into the RPV.

C. SLC will not inject into the RPV because the SLC PUMP B should have started when the key was turned to RUN position.

D. SLC will not inject into the RPV because the SQUIB Valve should have fired when the key was turned to the RUN position.

Answer:	D

Explanation:

SLC will not inject because the SQUIB Valve did not fire, as indicated by the white light energized.

Per R-STM-0201, Standby Liquid Control System, the SLC system is actuated with one of the two key locked SLC pump control switches on H13-P601. Each switch controls one of the 100% capacity subsystems. Upon placing the switch to RUN the following actions take place:

1. Squib valve fires, as indicated by the extinguishing of the white squib continuity light, the illumination of the loss of continuity status light and the initiation of the SLC out of service lights.
2. The SLC pump suction valve opens, as indicated by the red open indication light.
3. The Reactor Water Cleanup System isolation valve shuts to prevent removal of the poison solution from the reactor vessel.
4. The SLC pump starts, after the suction valve strokes full open, as indicated by the red pump running light above the pump control switch.
5. Successful SLC subsystem initiation is indicated by SLC pump start indication, SLC pump suction valve open indication, squib valve continuity lost discharge header pressure increasing to reactor pressure, SLC storage tank level lowering.

Distracters:

A. Plausible if applicant confuses initiation sequence.

B. Plausible if applicant confuses white continuity light with open indication or actual SQUIB valve firing.

C. Plausible if applicant confuses initiation sequence.

K/A Match

Applicant is required to understand the indications of the SQUIB Valve and the effect on the system initiation sequence when fault exists.

Technical References:

R-STM-0201, Standby Liquid Control System, Revision 9

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective: RLP-STM-0201-LO, Stanby Liquid Control, Objective (4): Describe how the following controls, indications, and/or interlocks affect the status of the SLC System. (4) d) Squib Valves		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(6)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="6">212000 Reactor Protection System
K3. Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following:
K3.05 RPS logic channels</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>212000 K3.05</td></tr>
<tr><td>Rating</td><td>3.7</td></tr>
<tr><td>Revision</td><td>2</td></tr>
<tr><td colspan="2"></td></tr>
<tr><td colspan="3">Revision Statement:
2: Corrected stem “18E.”</td></tr>
</table>

Question: 8

I&C is performing maintenance which inserts a half scram on RPS A.

While performing the maintenance, a fuse is blown (F18E) that de-energizes RPS Div 1 Scram SOV Valve CR1B.

Approximately how many rods should be inserted?

A. None

B. ½ of the rods

C. ¼ of the rods

D. All of the rods

Answer: C
Explanation: Per R-STM-0508, Reactor Protection System, The RPS is comprised of two trip systems, A and B, each made up of two trip logic channels (FIGURE 1). Trip Logic Channels A and C make up Trip System A while Trip System B consists of Channels B and D. The RPS logic arrangement is referred to as a, “one-out-of-two, taken twice” design. A trip in either of the two channels in Trip System A, coincident with a trip from either channel in Trip System B will result in a reactor scram. Note that a trip in both Trip Systems is required for a scram (what is termed a “Full Scram”). One or more trip conditions detected by the channels in one of the trip systems, without a trip condition detected in the other trip system, will not result in a scram. This condition is commonly referred to as a “Half Scram.” Per R-STM-508, Figure 7 Scram Pilot Valve Logic, the half scram will deenergize the A

solenoids for each RPS division. The blown fuse will deenergize the B solenoid for Division 1 only. The malfunction will insert the Group 1 rods (1/4 of the total rods).

Distracters:

A. Plausible if applicant only recognizes the half scram indications and does not understand the impact of the blown fuse. No rods are inserted on a half scram only.

B. Plausible if applicant confuses the RPS logic requirements with a half scram indication and blown fuse which insert half of the control rods.

D. Plausible if applicant recognizes the half scram indications and misdiagnoses the fuse failure as a failure in the other trip channel which inserts a full scram..

K/A Match

The applicant must have knowledge of RPS logic requirements to understand the impact of the blown fuse and half scram.

Technical References:

R-STM-0508, Reactor Protection System, Revision 8

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0508-LO, Objectives (12): Given a set of plant conditions, predict the effect that a loss or malfunction of the Reactor Protection System will have on the following: (12)
RPS Logic Channels

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X

10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Risk Significant Systems: Reactor Protection System		

Examination Outline Cross Reference	Level	RO
212000 Reactor Protection System A4. Ability to manually operate and/or monitor in the control room: A4.07 System status lights and alarms	Tier	2
	Group #	1
	K/A	212000 A4.07
	Rating	4.0
	Revision	2
Revision Statement: 2: Re-worded stem and changed distractor B to be more balanced (1 AND 3 instead of 2 AND 3).		

Question: 9

Based on the indications below:

Which RPS division(s) AT A MINIMUM must be manually actuated to achieve a full RPS scram?

A. RPS DIV 1 OR 3 (Only 1 required)

B. RPS DIV 1 AND 3 (Both required)

C. RPS DIV 2 OR 4 (Only 1 required)

D. RPS DIV 2 AND 4 (Both required)

Answer: C
Explanation: NOTE: The photos are arranged in the same order they would appear on the P680 from left to right. Per R-STM-0508, Reactor Protection System, The RPS is comprised of two trip

systems, A and B, each made up of two trip logic channels (FIGURE 1). Trip Logic Channels A and C make up Trip System A while Trip System B consists of Channels B and D.

The RPS logic arrangement is referred to as a, “one-out-of-two, taken twice” design. A trip in either of the two channels in Trip System A, coincident with a trip from either channel in Trip System B will result in a reactor scram. Note that a trip in both Trip Systems is required for a scram (what is termed a “Full Scram”).

As shown in the photos, the A trip system has been actuated. To insert a full scram, trip system B will need to be actuated. The B trip system can be actuated using either channel s B or D.

Distracters:

A. Plausible if applicant confuses with needing to actuate A trip system using A or C.

B. Plausible if applicant confuses with MSIV scram logic.

D. Plausible if applicant confuses with needing to trip both channels in Trip System B.

K/A Match

Applicant must interpret RPS light indications and apply to determine remaining logic required to manually insert a full scram.

Technical References:

R-STM-0508, Reactor Protection System, Revision 8

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0508-LO, Objectives (5): Describe how the following controls, indications, and/or interlocks affect the status of the Reactor Protection System: (5)
Manual Scram pushbuttons

Question Source:	**Bank #**	
	Modified Bank #	
	New	X

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Risk Significant Systems: Reactor Protection System		

Examination Outline Cross Reference	Level	RO
215003 Intermediate Range Monitor (IRM) System K2. Knowledge of electrical power supplies to the following: K2.01 IRM channels/detectors	Tier	2
	Group #	1
	K/A	215003 K2.01
	Rating	2.5
	Revision	1
Revision Statement:		

Question: 10

IRM 'F' detector is powered by ____________.

A. RPS A

B. RPS B

C. NHS-MCC2E

D. VBN-PNL01B1

Answer: B

Explanation:

RPS B supplies power to IRMs B, D, F, & H

Distracters:

A. RPS A supplies power to IRMs A, C, E, & G.
C. NHS-MCC2E supplies power to the IRM drive mechanism, but not the detectors.
D. VBN-PNL01B1 supplies the IRM recorders.

K/A Match

Applicant must have knowledge of power supplies for IRM F.

Technical References:

R-STM-0503

Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0503 Objective. 37: Given a set of plant conditions, predict the consequences a loss or malfunction of the following will have on the IRM System: (37) Reactor Protection System (RPS)		
Question Source:	**Bank # Dec 2010 NRC Q35**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(2)	
Level of Difficulty:	2	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
215004 Source Range Monitor (SRM) System K1. Knowledge of the physical connections and/or cause-effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and the following: K1.06 Reactor vessel	Tier	2
	Group #	1
	K/A	215004 K1.06
	Rating	2.8
	Revision	2
Revision Statement: 2: Cleaned up distractors B, C, and D.		

Question: 11

During a reactor startup SRM detectors are:

A. maintained in the fixed location in core for the life of the core.

B. fully withdrawn from the core when all IRMs are > 5 on Range 1.

C. incrementally withdrawn from the core maintaining 10^3 to 10^5 cps.

D. fully withdrawn from the core when the first IRM is on Range 2.

Answer: C

Explanation:

During a reactor startup, SRM detectors are gradually withdrawn from the core. This withdrawal causes the period meter to move in the negative direction.

Distracters:

A. Plausible because this describes APRM operation.
B. Plausible because this describes IRM operation.
D. Plausible because SRMs are permitted to be withdrawn when both of the IRMs associated with the SRM are on range 3 or above.

K/A Match

Applicant must have knowledge of SRMs physical position during a reactor startup.

Technical References: R-STM-0503, Rev 9		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0503-LO, Objective (3): Describe the SRM detectors full-in and full-out position with respect to the reactor core. (3)		
Question Source:	**Bank # GGNS 2015 NRC #35**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(2)	
Level of Difficulty:	2	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
215005 Average Power Range Monitor/Local Power Range Monitor System K3. Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: K3.04 Rod control and information system	Tier	2
	Group #	1
	K/A	215005 K3.04
	Rating	3.4
	Revision	2
Revision Statement: Revised stem to “Subsequently, a malfunction of APRM A results in the indications shown		

Question: 12

A sequence exchange is in progress.

APRM E has failed and all actions have been completed to continue the sequence exchange.

Subsequently, a malfunction of APRM A results in the indications shown.

What is the result of the indications?

A. Half scram actuated. APRM A ***can*** be bypassed and sequence exchange continued.

B. Half scram actuated. APRM A ***cannot*** be bypassed to continue sequence exchange.

C. Rod block actuated. APRM A ***can*** be bypassed and sequence exchange continued.

D. Rod block actuated. APRM A ***cannot*** be bypassed to continue sequence exchange.

Answer: D
Explanation: Per R-STM-0503 Neutron Monitoring, a control rod block occurs at the APRM downscale (5%). The rod block cannot be bypassed to continue the startup. APRM E is bypassed based on the given indications. Both APRM A and E cannot be bypassed at the same time. The Division 1 joystick only bypasses one APRM at a time. Division 1 Joystick can only bypass APRM Channel A, C, E, OR G.
Distracters: Bypassing APRM Channel A is plausible if applicant confuses indications and joystick operation. Division 1 Joystick can only bypass APRM Channel A, C, E, OR G. There is a Division II Joystick for Channels B, D, F, OR H. Half scram is plausible if applicant confuses indications between APRM A and E. Also if the applicant confuses the setpoints for a half scram and rod block. The APRM E upscale will actuate a half scram.
K/A Match Applicant must interpret indications for APRM A for downscale indications. Then the applicant must apply that downscale indication to rod block setpoint. The rod block prevents the RC&IS system from moving rods further in the startup until the condition is cleared.
Technical References: R-STM-0503 Neutron Monitoring, Revision 11
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: RLP-STM-503, Nulcear Instrumentation, Objective 24 and 25: List the signals and setpoints, which results in the initiation of a SCRAM and Rod Block. (24) Given the following H13-PNLP680 APRM Annunciators, state the cause of the alarm

and status of the APRM System: (25) APRM Upscale Trip or INOP APRM Upscale APRM Downscale		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(2)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	RO
217000 Reactor Core Isolation Cooling System (RCIC) K2. Knowledge of electrical power supplies to the following: K2.02 RCIC initiation signals (logic)	**Tier**	2
	Group #	1
	K/A	217000 K2.02
	Rating	2.8
	Revision	3
Revision Statement: Revision 2: Replaced 120 VAC and 125 VDC with 2 alternating VAC and VDC bus designators Revision 3: Added “bus” to distractors C and D. Also added information from STM to explanation.		

Question: 13

RCIC relay logic is powered from ___(1)___ and on a loss of that power RCIC will ___(2)___.

	(1)	(2)
A.	ENB-PNL02A and ENB-PNL02B	actuate
B.	ENB-PNL02A and ENB-PNL02B	not respond
C.	RPS BUS A and RPS BUS B	actuate
D.	RPS BUS A and RPS BUS B	not respond

Answer: B

Explanation:

Per R-STM-209, RCIC, relay logic and essential valves for the RCIC system are powered from 125VDC ENB-PNL02A and ENB-PNL02B. On a loss of DC RCIC will not respond.

Distracters:

120VAC is plausible if applicant confuses with power for RPS Buses.

Also, if applicant confuses actuation with RPS, on a loss of power RPS will actuate.

In addition to supplying the RPS Trip Systems, the RPS buses supply power to the Nuclear Steam Supply Shutoff System, the Neutron Monitoring System, and the two Process Radiation Monitoring System Main Steam Line Monitor channels.

K/A Match Applicant must have knowledge of the power that supplies the RCIC initiation logic and what is the effect if the power is lost.		
Technical References: R-STM-209, RCIC, Revision 13 R-STM-508, Reactor Protection System, Revision 8		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-209-LO, Objective (13): Describe the interrelationship(s) between the following systems and the RCIC System: (13) AC Electrical Distribution DC Electrical Distribution		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(8)	
Level of Difficulty:	2	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Risk Significant Systems: Safety Related DC Power		

Examination Outline Cross Reference	Level	RO
218000 Automatic Depressurization System K6. Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM : K6.04 Air supply to ADS valves:	Tier	2
	Group #	1
	K/A	218000 K6.04
	Rating	3.6
	Revision	2
Revision Statement: Centered the (1) and (2) above the columns		

Question: 14

If a loss of offsite power occurs, the **initial** backup air supply for the SRVs is provided by the ___(1)___ and is ___(2)___ aligned.

NOTE: IAS: Instrument Air System
PVLCS: Penetration Valve Leakage Control System Air Supply System

	(1)	(2)
A.	IAS Diesel Air Compressor	automatically
B.	PVLCS Compressors	manually
C.	IAS Diesel Air Compressor	manually
D.	PVLCS Compressors	automatically

Answer: D

Explanation:

Per R-STM-0202, ADS, PVLCS provides a backup source of air for ADS valve operation. During a loss of offsite power event, the normal air supply for the SRVs (SVV compressors) will be de-energized. The PVLCS compressors which are powered from the emergency diesel generators via safety related buses will serve to provide air for SRV operation. No operator action is required to align PVLCS to SVV.

Distracters:

The IAS Diesel Air Compressor is capable of supplying air to SRVs, but it is not the initial backup air supply.

IAS Diesel compressor is plausible because the plant air system has a diesel driven air

compressor. The diesel driven air compressor provides a source of air to the Instrument Air System and Service Air System in case of a loss of offsite power (LOP). This compressor must be manually connected , aligned, and started to supply SVV. See SOP-0011 Main Steam for how the IAS Diesel Air Compressor is aligned to SVV		
K/A Match Applicant must have knowledge of what the effect of a loss of SVV air will have on the ADS valves and what is the backup source of air and the method to place it on service.		
Technical References: R-STM-0202, Automatic Depressurization System R-STM-0121, Plant Air Systems		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0202-LO, Objective 12: Given a set of plant conditions, predict the consequences of a loss or malfunction of the following will have on the Automatic Depressurization System : (12) Air supply to ADS valves		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(3)	
Level of Difficulty:	2	
SRO Only Justification: N/A		

PRA Applicability:
Top 10 Risk Significant Systems: SRV Air Compressor (SVV)

Examination Outline Cross Reference	**Level**	RO
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off K4. Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: K4.01 Redundancy	**Tier**	2
	Group #	1
	K/A	223002 K4.01
	Rating	3.0
	Revision	2
Revision Statement: Replaced Distractor C with "ONLY the MSIVs are CLOSED." Split up stem as specified, used modifier "inadvertently" for "mistakenly".]		

Question: 15

- The plant is in Mode 1.
- I&C is performing STP-058-4201, Containment and Drywell Manual Isolation Actuation LSFT.
- The Reactor Operator arms and depresses the "A" CRVICS pushbutton on H13-P680, while I&C continues with verifications per the STP.
- Prior to the isolation signal being reset, the "D" CRVICS pushbutton is inadvertently armed and depressed.

Which of the following is the status of the MSIVs and Containment Isolation Valves?

A. ONLY the Main Steam Isolation Valves (MSIVs) and Outboard MSL Drains are CLOSED.

B. The MSIVs, and Outboard MSL Drains <u>AND</u> the Outboard BOP Containment Isolation Valves are CLOSED.

C. ONLY the MSIVs are CLOSED.

D. ONLY the Outboard BOP Containment Isolation Valves are CLOSED.

Answer: A
Explanation: The design of the Containment and Reactor Vessel Isolation Control System (CRVICS)

is such that the only the MSIV logic channels seal in when using the CRVICS pushbuttons. Since the A then D sequentially pressed, MSIV logic channels A and D actuated and sealed in. This combination closes all MSIVs. However, A and D MSIV logic channels sealed in only closes the outboard MSL drains signals seal in, but the BOP isolations do not; therefore pushing the pushbuttons sequentially will only cause the MSIVs and the outboard MSL Drains to close.

Distracters:

B. Incorrect, Although the MSIVs, and Outboard MSL Drains would close, the outboard BOP isolation valves would not close. The BOP isolation logic channels do not seal-in until an isolation signal is generated (for example pressing A & D simultaneously would close the outboard BOP isolation valves) Plausible if the applicant misunderstands the system design and believe that the individual BOP isolations logic channels seal-in . Additionally, A and D pushbuttons are associated with outboard BOP isolation valves. B and C pushbuttons are associated with the inboard BOP isolation valves.

C. Incorrect, The BOP isolation logic channels do not seal-in until an isolation signal is generated (for example pressing A & D simultaneously would close the outboard BOP isolation valves) Plausible if the applicant misunderstands the system design and believe that the individual BOP isolations logic channels seal-in . Additionally, A and D pushbuttons are associated with outboard BOP isolation valves. B and C pushbuttons are associated with the inboard BOP isolation valves.

D. Incorrect, The BOP isolation logic channels do not seal-in until an isolation signal is generated (for example pressing A & D simultaneously would close the outboard BOP isolation valves) Plausible if the applicant misunderstands the system design and believe that the individual BOP isolations logic channels seal-in .

K/A Match

The applicant must have the required knowledge of the pushbutton operation and the effects on the Containment and Reactor Vessel Isolation Control System. Also, the applicant must understand which signals are sealed in and what effect sequentially operating isolation pushbutton will have.

Technical References:

R-STM-0058, CRVICS, Rev 9 p. 39 of 63

Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0058, Obj 11: Describe the CRVICS design features which provide for the following: (11) Redundancy		
Question Source:	**Bank # DEC 2014 NRC Q43**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(3)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Events: MSIV Closure		

Examination Outline Cross Reference	**Level**	RO
239002 Relief/Safety Valves A1. Ability to predict and/or monitor changes in parameters associated with operating the RELIEF/SAFETY VALVES controls including: A1.08 Suppression pool water temperature	**Tier**	2
	Group #	1
	K/A	239002 A1.08
	Rating	3.8
	Revision	1
Revision Statement: Revised distractors to delete 120°F. Used the limit for containment air temperature which is 95° as a plausible distractor for A. Revised explanations to support the question revision. Revised stem to include referenced TS		

Question: 16

While operating at 100% power, an SRV inadvertently opens.

The SRV must be manually closed prior to what maximum average suppression pool temperature, to prevent entering technical specification 3.6.2.1, Suppression Pool Average Temperature?

A. 95°F

B. 100°F

C. 105°F

D. 110°F

Answer: B

Explanation:

Per Tech Spec 3.6.2.1, Suppression Pool Average Temperature, suppression pool average temperature shall be ≤100°F when THERMAL POWER IS >1% RTP and no testing that adds heat to the suppression pool is being performed. The plant is operating at 100% and there is to testing in progress adding heat to the pool.

Distracters:

A. 95°F is plausible because this is the limit for containment air temperature. The applicant could confuse suppression pool temperature limit with containment air temperature limit.

C. 105°F is plausible if applicant does not understand testing adding heat to the pool. 105°F is the limit when testing is being performed.

D. 110°F is plausible if applicant does not remember the power requirement. 110°F is allowed when power is ≤1%.

K/A Match

Applicant must have knowledge of suppression pool temperature limits when heat is being added to the pool to ensure the open SRV is manually closed prior to the technical specification temperature limit.

Technical References:

Tech Spec 3.6.2.1, Suppression Pool Average Temperature

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0057-LO Objective 9: Identify the technical specifications and/or technical requirements manual requirements related to primary containment. (9)

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(8)	
Level of Difficulty:	3	

SRO Only Justification:

N/A

PRA Applicability:

Examination Outline Cross Reference	Level	RO
259002 Reactor Water Level Control K5. Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM : K5.03 Water level measurement	Tier	2
	Group #	1
	K/A	259002 K5.03
	Rating	3.1/3.2
	Revision	0
Revision Statement: Rev 0: New KA selected by NRC for Q17.		

Question: 17

The unit is operating at 65% rated power with plant conditions as follows:

- Narrow Range Water Level N004C has failed upscale high.
- Narrow Range Water Level N004B is selected for input to Feedwater Level Control System.

Which of the following describes the expected response if the Narrow Range Water Level N004B fails upscale high?

A. The Feedwater Regulating Valves will respond to maintain RPV level.

B. The Feedwater Regulating Valves will close ONLY

C. The Reactor Feedwater Pumps will trip and the Feedwater Regulating Valves will close.

D. The Reactor Feedwater Pumps will trip and the Feedwater Regulating Valves will open.

Answer: C

Explanation:

The Reactor Feedwater Pump high RPV level trip logic is 2 out of 3 level instruments greater than Level 8. When the 2nd narrow range level instrument fails high, the reactor feed pumps will trip. The input to the feedwater level control system is manually selected by RX LVL A and RX LVL B pushbuttons and will not automatically swap when level instrument failures are sensed by signal failure circuitry (RX FW LEVEL CONTROL SIGNAL FAILURE ARP P680-03A-C08). When the Narrow Range B instrument fails upscale, since it is selected for input, they feedwater regulating valves will close due to sensing a false high RPV level condition. Therefore, when N004B fails HIGH, the reactor feed pumps will trip and the feed regulating valves will close.

Distracters:

A. Plausible if an applicant is not familiar with the Feedwater Level Control System input selection circuitry and the purpose of the RX FW LEVEL CONTROL SIGNAL FAILURE alarm. (Detects greater than a six-inch mismatch between level instruments and has no inherent automatic functions.) Also, applicant may not be familiar with the logic arrangement for feed pump high level trip circuitry or may believe that a gross failure of the level instrument will prevent a RFP trip.

B. Plausible if applicant is not familiar with the logic arrangement for feed pump high level trip circuitry, but is familiar with the Feedwater Level Control System input selection circuitry.

D. Plausible if applicant is not familiar with the Feedwater Level Control System input selection circuitry, but is familiar with the logic arrangement for feed pump high level trip circuitry. Additionally, if the applicant recognizes that the reactor feed pumps trip resulting in RPV level lowering that the feedwater regulating valves would respond by opening.

K/A Match

The operational implications of water level measurement on the Feedwater Level Control System is demonstrated in the question by a failure of multiple level instruments resulting in a plant transient.

Technical References:

R-STM-0107 Feedwater & Feedwater Level Control Systems Rev 32

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

STM-107B OBJ-H8

QuestioSource:	**Bank . Question ID: 646290**	X
(note changes and attach parent)	**Modified Bank #**	
	New	

Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	

10CFR Part 55 Content:	41(b)(2)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		
N/A		

Examination Outline Cross Reference	Level	RO
Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: A3.01 System flow	Tier	2
	Group #	1
	K/A	261000 A3.01
	Rating	3.2
	Revision	1
Revision Statement:		

Question: 18

A reactor coolant leak has occurred in the Drywell.

- Drywell pressure is 2.14 psid and stable.
- Both trains of Standby Gas Treatment (GTS) started. OSP-0053 Hard Card Attachment 21, Operating Auxiliary Building Ventilation was used to reduce to one train of SBGT running; GTS-A is running.

Subsequently, a loss of RSS#1 occurred and the Div 1 Diesel Generator failed to start.

What is the status of GTS-B following the loss of RSS#1?

A. GTS-B may only be started in High Volume Purge mode due to power failure.

B. GTS-B must be manually initiated due to manually securing.

C. GTS-B will restart automatically due to low flow in GTS-A.

D. GTS-B will restart automatically due to undervoltage trip of GTS-A.

Answer: C

Explanation:

DW 1.68 is still locked in, so when low flow occurs in Div 1 due to the loss of power, then the Div 2 will automatically restart.

Distracters:

A. Incorrect - Power is only lost to Div 1; Div 2 is still available
B. Incorrect - Securing per the hard card places GTS-B in standby;

D. Incorrect - The interlock that will start GTS-B is low flow in the running GTS; not undervoltage.		
K/A Match The applicant must have knowledge of the power supplies to the standby gas treatment systems to describe the proper operating due to the partial loss of AC power. The applicant must also have the knowledge of the low flow auto-start feature to be able to verify proper operation.		
Technical References: R-STM-0257, Standby Gas Treatment, Rev 5 p. 15 of 28		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0257, Objective 12: Given a set of plant conditions, predict the consequences of a loss or malfunction of the following will have on the Standby Gas Treatment System (12) a) AC Electrical Distribution		
Question Source:	**Bank # Dec 2014 NRC Q46**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: N/A		

PRA Applicability:

Examination Outline Cross Reference	Level	RO
Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: K4.01 Bus lockouts	Tier	2
	Group #	1
	K/A	262001 K4.01
	Rating	3.0
	Revision	2

Revision Statement:
Edited the stem to say "… the Normal xfmr supply breaker __(1)__ be closed locally … ", and revised the correct answer as B. Revised explanation to support the change in the stem and a correct answer of B

Question: 19

LOCKOUT/RESET pushbuttons are provided on panel H13-P808 for the "Normal" and "Preferred" transformer supply breakers.

When the LOCKOUT pushbutton is depressed, the "Normal" transformer supply breaker ___(1)___ be closed locally, and the associated "Preferred" supply breaker ___(2)___ automatically close upon receipt of an auto bus transfer signal.

	(1)	(2)
A.	can	will
B.	can	will not
C.	cannot	will
D.	cannot	will not

Answer: B

Explanation:

LOCKOUT/RESET pushbuttons are provided on panel H13-P808 for the "Normal" and "Preferred" transformer supply breakers. When the LOCKOUT pushbutton is depressed, the associated breaker can be closed locally, but cannot be closed remotely, nor will an associated "Preferred" supply breaker close upon receipt of an auto bus transfer signal.

Distracters: A, C, D are all plausible distractors regarding the effect and interlock associated with the LOCKOUT pushbutton. Although the normal supply breaker can be closed locally, it cannot be closed remotely making C&D plausible. There are separate lockout pushbuttons for different supply breakers, so if the applicant confuses the “associated” breaker with any other breaker the auto bus transfer signal CAN occur.		
K/A Match The applicant must have knowledge of the interlock the lockout pushbutton provides on the associated breakers and AC distribution system.		
Technical References: R-STM-0300, AC Distribution, Rev 31		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-300-LO, AC Distribution, Objective 5: Given a labeled diagram/drawing of the A.C. Distribution control panel, H13-P808/680, identify/state. (5) The effect the manipulation of each control switch will have on AC Distribution.		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	

SRO Only Justification: N/A
PRA Applicability:
Top 10 Risk Significant Systems: NPS 13.2 kV AC Power

Examination Outline Cross Reference	**Level**	RO
Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.11 Degraded system voltages	**Tier**	2
	Group #	1
	K/A	262001 A2.11
	Rating	3.2
	Revision	2
Revision Statement: Shifted (1) and (2) in-line with the answers.		

Question: 20

The plant is operating at 100% power when alarm P808-86A, Grid Trouble is received. SPI-REC102 on P808 indicates 230 KV system voltage at Fancy Point to be 223.3 KV.

The System Operations Center (SOC) contacts the control room and indicates that a transient has occurred and Fancy Point is <u>NOT</u> sufficiently stable at this time.

(1) What is the impact to plant equipment?
(2) What actions should be taken to mitigate this event?

A. (1) Amps will increase on the running equipment.
   (2) Perform a normal start of the D/G, parallel it to offsite power and disconnect the bus from the grid.

B. (1) Amps will increase on the running equipment.
   (2) Perform an emergency start of the D/G and open the safety related bus supply breaker.

C. (1) Amps will decrease on the running equipment.
   (2) Perform an emergency start of the D/G and open the safety related bus supply breaker.

D. (1) Amps will decrease on the running equipment.
   (2) Perform a normal start of the D/G, parallel it to offsite power and disconnect the bus from the grid.

Answer: B
Explanation: (1) In a degraded grid condition, the 230 KV voltage supplied through transformers to in plant loads results in reduced bus voltages. With motors power requirements constant (based on supply frequency) will result in higher amperages bus voltage degrades.

(2) Correct per AOP-64 Section 5.6.3 which states that if Fancy Point is not sufficiently stable, then the diesel generators are started in the emergency mode, then each safety bus is de-energized by opening the supply breaker and the DG output breaker automatically closes. This method is required since with unstable conditions, the DGs cannot be reliably paralleled to the offsite power source.

Distracters:

A. (1) Correct

(2) Incorrect. Would be correct if Fancy Point was stable. When grid conditions are sufficiently stable, it allows DGs to be normally started and paralleled to their respective buses prior to disconnecting from offsite power. Plausible if an applicant does not understand the interrelationship between grid stability and transferring buses to the DGs

C. (1) Incorrect but plausible if the applicant is not aware that the majority of in plant loads consist of rotating equipment and him and him and not resistive loads. If the majority of in plant equipment was resistive loads then amperage with lower as supply voltage lowered.

(2) Correct

D. (1) incorrect as described in C

(2) incorrect as described in A

K/A Match

Test item describes an unstable degraded grid condition. The applicant requires knowledge of the impacts of degraded voltage on running equipment and procedure actions necessary to restore safety bus voltages to normal.

Technical References:

AOP-0064, Degraded Grid, Rev 6 p 8 of 11

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-OPS-AOP064, Objective 4: D. Analyze the operational impact of the related

operational experiences associated with the Degraded Grid. (4)		
Question Source:	**Bank # Dec 2014 NRC Q47**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Risk Significant Systems: 230 kV AC Power		

Examination Outline Cross Reference	Level	RO
Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: A3.01 Transfer from preferred to alternate source	Tier	2
	Group #	1
	K/A	262002 A3.01
	Rating	2.8
	Revision	2
Revision Statement: Revised stem to “This condition will be indicated by _____ “]		

Question: 21

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

This condition will be indicated by _____________

A. annunciator in the main control room AND status light indication at the inverter.

B. annunciator in the main control room AND status light indication on H13-P808.

C. ONLY status light indication at the inverter.

D. status light indication at the inverter AND status light indication on H13-P808.

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.

Distracters:

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.

K/A Match The applicant must know what indications to monitor when there is a change in the power supply to the inverter from the preferred to the alternate power source.		
Technical References: R-STM-0300 Rev 26 Figure 7 ; ARP-808-87A, Rev 24 Pg 10 of 50		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0300, Objective 16: Given a set of plant conditions, predict the consequences a loss or malfunction of the following will have on Uninterruptible Power Supply (AC/DC): (16) AC Power		
Question Source:	**Bank # Mar 2014 NRC Q47**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
263000 D.C. Electrical Distribution 2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.	Tier	2
	Group #	1
	K/A	263000 2.4.45
	Rating	4.1
	Revision	1
Revision Statement:		

Question: 22

Which one of the following annunciators has one or more expected automatic actions?

A. EHC 125VDC POWER LOST

B. DIV III 125VDC SYSTEM TROUBLE

C. DIV I 125VDC PWR LOSS/INPUT CARD P951A OOF

D. 125VDC ISOLATOR POWER LOSS OR CARD OUT OF FILE

Answer: A

Explanation:

Per H13-P870/54A/H03, EHC 125VDC POWER LOST, the associated automatic action on a loss of 125VDC power Main Turbine trips.

Distracters:

B. H13-P601/16A/E02, DIV III 125VDC SYSTEM TROUBLE is plausible if applicant believes Div III trouble will initiate HPCS.
C. H13-P680 / 03A / A14, DIV I 125VDC PWR LOSS/INPUT CARD P951A OOF is plausible if applicant believes Div 1 loss of power will initiate division 1 ESF equipment.
D. Plausible if applicant misunderstands H13-P601/16A/F04, 125VDC ISOLATOR POWER LOSS OR CARD OUT OF FILE, and believes loss of isolator power causes an automatic action.

K/A Match

Applicants must be able to recognize which alarm procedure has an associated automatic action.		
Technical References: H13-P870/54A/H03, EHC 125VDC POWER LOST H13-P601/16A/E02, DIV III 125VDC SYSTEM TROUBLE H13-P680 / 03A / A14, DIV I 125VDC PWR LOSS/INPUT CARD P951A OOF H13-P601/16A/F04, 125VDC ISOLATOR POWER LOSS OR CARD OUT OF FILE		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0305-LO Objective 14: Given the ARP, state the expected indication, the impact to the DC Distribution System, and required actions for each valid annunciator alarm. (14)		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(10)	
Level of Difficulty:	4	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
264000 Emergency Generators (Diesel/Jet) A4. Ability to manually operate and/or monitor in the control room: A4.03 Transfer of emergency control between manual and automatic	Tier	2
	Group #	1
	K/A	264000 A4.03
	Rating	3.2
	Revision	1
Revision Statement:		

Question: 23

When the STBY DIESEL ENGINE EMERGENCY START pushbutton is manually depressed, in addition to an Overspeed trip, which of the following automatic shutdown trips are active:

A. All safety shutdown trips are active

B. Generator Differential Overcurrent ONLY

C. High Jacket Water Temp and High Lube Oil Temp ONLY

D. Generator Differential Overcurrent, High Jacket Water Temp, and High Lube Oil Temp ONLY

Answer: D
Explanation: Per R-STM-309S, Standby Diesel Generator, when the STBY DIESEL ENGINE EMERGENCY START pushbutton is depressed, all safety shutdown trips will be inhibited except Diesel Overspeed , Generator Differential Overcurrent. “High Jacket Water Temp” and “High Lube Oil Temp” are active trips on LOP or Manual Emergency Start.
Distracters: A. Plausible if applicant confuses this start with normal manual start. B. Plausible if applicant confuses this start with a LOCA start. C. Plausible if applicant confuses the additional active trips with LOP or Manual Emergency Start.
K/A Match The applicant must have knowledge of active automatic trips when the diesel generator

is started manually using the STBY DIESEL ENGINE EMERGENCY START pushbutton.
Technical References: R-STM-309S, Standby Diesel Generator
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: RLP-STM-0309S-LO Objective 5: Describe the following controls, indications and/or interlocks for the following: (5) Standby Diesel Generator (Auto Start Signals and Trip Signals)

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(7)&(8)	
Level of Difficulty:	3	

SRO Only Justification: N/A

PRA Applicability:
Top 10 Operator Actions: Recover a Diesel Generator within 1 hour

Examination Outline Cross Reference	Level	RO
300000 Instrument Air System (IAS) A3. Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including: A3.02 Air temperature	Tier	2
	Group #	1
	K/A	300000 A3.02
	Rating	2.9
	Revision	2
Revision Statement: Made the distractors equal to the setpoint.		

Question: 24

Which of the following conditions will cause an automatic trip of a running IAS compressor?

A. Discharge Air Moisture High -40°F Dew Point

B. Low System Air Pressure, 104 psig

C. IAS Header Pressure of 113 psig

D. High LP Air Outlet Temperature 450°F

Answer: D

Explanation:

Per R-STM-0121, Table 3 Setpoints and Interlocks, the IAS/SAS Compressor Trip setpoint for high LP air out temp is ≥450°F and only the affected compressor trips.

Distracters:

A. Plausible if applicant confuses setpoint for IAS Dryer Prefilter After Filter Trouble. The setpoint is Dryer 2 discharge air moisture high -40°F Dew Point. This is an alarm only and does not trip the compressor.

B. Plausible if applicant confuses setpoint for Low Service Air Pressure System isolation setpoint. The setpoint is 104 psig. This does not trip the compressor.

C. Plausible if applicant confuses setpoint for IAS Dryer Purge Line Isolation Valve isolation setpoint. The setpoint is 113 psig. This does not trip the compressor.

K/A Match

Applicant must have knowledge of automatic trip of air compressors on high air temperature condition to ensure proper automatic operation as required.

Technical References:

R-STM-0121, Plant Air Systems, Rev 18

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0121-LO, Objective 5: List the signals, including setpoints that will trip the air compressors (SAS and IAS). (5)

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	

SRO Only Justification:

N/A

PRA Applicability:

Top 10 Risk Significant Systems: Instrument and Service Air

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="6">400000 Component Cooling Water System (CCWS)
A2. Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
A2.02 High/low surge tank level</td><td>Tier</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>400000 A2.02</td></tr>
<tr><td>Rating</td><td>2.8</td></tr>
<tr><td>Revision</td><td rowspan="2">2</td></tr>
<tr><td></td></tr>
<tr><td colspan="3">Revision Statement:
Revised (2) to read “In accordance with “the associated” alarm response procedure, what action “should” be taken….?”</td></tr>
</table>

Question: 25

The plant is operating at 100% rated power.
An alarm was received in the Main Control Room:

RPCCW SURGE TANK TK1 EXTREME LOW LEVEL (H13-P870/55A/E04)

Local investigation revealed a broken air supply line to the tank makeup valve.
RPCCW surge tank level is just below the alarm setpoint and steady.

(1) What is the concern of reduced surge tank level?

(2) In accordance with the associated alarm response procedure, what action should be taken to mitigate this failure to allow continued operation?

A. (1) Loss of surge volume for thermal expansion.
(2) Align Service Water to supply CCP safety related loads.

B. (1) Reduction in NPSH to CCP pumps.
(2) Open manual bypass around failed AOV to make up to tank.

C. (1) Loss of surge volume for thermal expansion.
(2) Open manual bypass around failed AOV to make up to tank.

D. (1) Reduction in NPSH to CCP pumps.
(2) Align Service Water to supply CCP safety related loads.

Answer:	**B**

Explanation:

Per H13-P870/55A/E04, RPCCW SURGE TANK TK1 EXTREME LOW LEVEL, Long Term Actions, If the Level Control Valve fails to open, THEN maintain surge tank level using MWS-V238 LEVEL CONTROL VALVE BYPASS.

Loss of surge tank level results in decrease in NPSH for the CCP pumps.

Distracters:

Plausible if applicant confuses lower water level with less expansion volume. There is more volume in the surge tank for thermal expansion due to the lower water level.

Plausible if applicant confuses AOP-11 (Loss of Reactor Plant Component Cooling Water) actions. AOP-11, gives directions to align SSW to the CRD Pump Bearing Cooler and to the in-service Fuel Pool Cooling Heat Exchanger. Aligning Service Water does not supply all loads therefore continued operation would not be possible. (Example: Reactor Recirculation pumps would lose cooling).

K/A Match

Given main control room alarm, applicant must predict the impact of the low surge tank level alarm and determine which action must be taken to mitigate the low surge tank level per the alarm response procedure.

Technical References:

ARP-P870-55A-E04

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

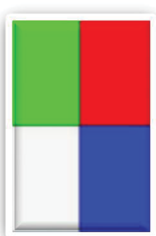
RLP-STM-0115 Objective 3: Describe the function, operation, control, indications, and/or interlocks of the following Reactor Plant Component Cooling Water components and/or subsystems: (3)
CCP Surge Tank

Question Source:	**Bank # April 2010 NRC Q53**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41(b)(4), 55.41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
510000 Service Water A1. Ability to predict and / or monitor changes in parameters associated with operating the Service Water controls including: A1.03 Service Water Pressure	**Tier**	2
	Group #	1
	K/A	510000 A1.03
	Rating	2.7
	Revision	2
Revision Statement: Revised K/A to “Service Water” and “Service Water Pressure” instead of “CCWS” and “CCW Pressure” Edited first question to read (1) When the Normal Service Water (NSW) pump has auto-tripped, which light is flashing? Included in part (2) distractors A&B low discharge pressure setpoint 97 psig. Added a value for low suction pressure trip setpoint of 5 psig to part (2) distractors C&D for consistency, Verified that it did not cue another question’s answer. Updated explanation with additional info for suction pressure trip setpoint and characteristics. Updated question references by highlighting info regarding suction pressure trip setpoint and characteristics**.**		

Question: 26

(1) When the Normal Service Water (NSW) pump has auto tripped, which light is flashing?



(2) Which NSW system parameter will cause the automatic start of the standby pump?

	(1)	(2)
A.	BLUE	Low pump discharge pressure of 97 psig.
B.	GREEN	Low pump discharge pressure of 97 psig.
C.	BLUE	Low pump suction pressure trip of 5 psig for running pump.
D.	GREEN	Low pump suction pressure trip of 5 psig for running pump.

Answer: B

Explanation:

Per STM-118, Service Water Systems, the GREEN light flashes on auto-trip or pump is stopped locally.

The Normal Service Water pump in STANDBY will automatically start with the Control Transfer Switch at the applicable NNS switchgear in REMOTE and the Harris Panel lockout reset or LOCAL AUTO and either an automatic trip of either running pump (except low suction pressure trip) OR a low service water pump discharge pressure of 97 psig.

Distracters:

The BLUE light flashes on breaker inoperative or if transfer switch is in either of the two LOCAL positions.

An automatic trip of either running pump is plausible if the applicant forgets the exception to the auto start feature. An automatic trip of either running pump will auto start the standby pump except for the low suction pressure trip. The low suction pressure trip of 5 psig (delayed 45 seconds). This condition will trip all running pumps and lockout any idle pumps.

K/A Match

Applicant must have knowledge of NSW pump operating indications and predict automatic pump start when the pump discharge pressure is <97 psig.

The normal service water system was added to the selection due to the safety significance of the system to the plant. The Normal Service Water system is designed in accordance with the following requirements. It provides cooling water to the secondary side of the Reactor Plant Component Cooling Water, Turbine Plant Component Cooling Water, and Plant Chilled Water (HVN) systems during normal plant operation and planned unit outages. It is also designed to supply cooling water to the Residual Heat Removal heat exchangers to dissipate reactor decay heat when the Standby Service Water system is not in use. The Normal Service Water system components are designed in accordance with the safety classifications listed in RBS USAR Table 3.2-1, Equipment and Structure Classification. The Normal Service Water system is designed to remove the heat load listed in RBS USAR NSW Major Components Design Data.

Technical References:

Per STM-118, Service Water Systems

Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-118, Objective 4: Given a labeled diagram/drawing of the Normal Service Water system control panel (Harris Panel), identify: (4) The function of each indicator The condition(s) which will cause the indicator to light or extinguish		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(4)	
Level of Difficulty:	4	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
201003 Control Rod and Drive Mechanism 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	Tier	2
	Group #	2
	K/A	201003 2.1.7
	Rating	4.4
	Revision	2
Revision Statement: Clarified in explanation why the rod does not settle at 00 after continuous insert to full-in, and the significance of that indication		

Question: 27

The reactor is operating at 30% power when the Reactor Operator identifies Control Rod 48-41 drifting in.

Drive water pressure is 250 psid and steady

A continuous insert signal is applied to the drifting rod until it is full-in. When the continuous insert signal is removed, the rod does not settle to position “00”.

Which of the following identifies the cause of Control Rod 48-41 drifting in?

A. CRD drive pressure too high

B. Scram outlet valve is partially open

C. Stuck Collet piston

D. Control rod is uncoupled

Answer: B

Explanation:

For the conditions given in the stem the operator has indication that 1 control rod is drifting into the core. With the scram outlet valve partially open, the differential pressure on the control rod drive piston is sufficient to cause the control rod to drift in at a rate proportional to the position of the scram outlet valve. The differential pressure on the CRDM drive piston is created because the scram outlet valve vents the CRDM above piston area to the scram discharge volume which is at atmospheric pressure and the below piston area is supplied by reactor pressure through the scram ports located on the CRDM flange. (See Rod Drift Alarm)

Since the scram discharge volume vent and drain valves remain open since no scram signal is present, the scram discharge volume remains at atmospheric pressure. Since the differential pressure across the CRDM drive piston (causing the control rod to drift in) remains constant, the control rod to remain beyond the full in position. Therefore, when the continuous insert signal is removed, the control rod does not settle at the 00 position.

Distracters:

A While high drive water pressure could cause a control rod to move faster than normal, it will not cause the control rod to drift without another failure. In addition the pressure given in the stem is normal for the given plant conditions.(STM page 16)

C a stuck collet piston would cause a control rod to drift out of the core. The stuck piston would not allow the collet fingers to close around the index tube allowing the weight of the rod to push the mechanism out of the core. (See Rod Drift Alarm Q27)

D An uncoupled control is discovered when performing a coupling check. The indications for an uncoupled rod is that the control rod “drifts” past position 48 when a withdraw signal is given. The operator response is to insert the rod to attempt recoupling. An applicant can confuse the symptoms of the response of the rod to an insert signal and the rod remaining “beyond full in” to an uncoupled rod that remains “beyond full out” (See Rod Drift Alarm, Control Rod Overtravel alarm, and SOP-71 Coupling Check Procedure)

K/A Match

This question requires knowledge of control rod drive hydraulics and mechanism. The applicant applies this knowledge to determine a cause for a control drift based on symptoms and interpretation of control rod movement response.

Technical References:

STM-52 Control Rod Drive System
Rod Drift Alarm
Control Rod Overtravel alarm
SOP-71 Coupling Check Procedure

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective: RLP-STM-052-LO Objective 9: Identify / Describe the controls, indications, and Interlocks for the following: (9) Control Rod Drift		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	CFR 55.41(b)6	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	RO
201005 Rod Control and Information System (RCIS) K1. Knowledge of the physical connections and/or cause-effect relationships between ROD CONTROL AND INFORMATION SYSTEM (RCIS) and the following: K1.02 Reactor/turbine pressure control system: BWR-6	**Tier**	2
	Group #	2
	K/A	201005 K1.02
	Rating	3.3
	Revision	2
Revision Statement: Modified stem to: "Which of the following instrument failures would cause an increase in the allowable notches per rod withdrawal?"		

Question: 28

During a reactor power ascension, reactor power is at 75%.

The crew is preparing to withdraw control rods to the target rod pattern.

Which of the following instrument failures would cause an increase in the allowable notches per rod withdrawal?

A. Feedwater flow instrument fails downscale.

B. Feedwater flow instrument fails upscale.

C. Turbine 1st stage pressure instrument fails downscale.

D. Turbine 1st stage pressure instrument fails upscale

Answer: C
Explanation: When power level is between the LPSP (27.5%) and the High Power Setpoint (HPSP) (67.9%), the Rod Worth Limiter (RWL) limits allowable notches of withdrawal for any control rod to 4 notches. Once power level passes above the HPSP (again sensed by first stage turbine pressure) the RWL limits allowable notches of withdrawal for any control rod to 2 notches. If the turbine 1st stage pressure instrument fails downscale, then the RWL will sense power below the HPSP and allow for notches of control rod withdrawal which is non-conservative because it allows extra notches of withdrawal.

The question tests if applicant recognizes what instrument determines LPSP and HPSP affecting the RWL. Additionally, the applicant needs to evaluate what type of failure would cause the RWL to fail non-conservatively.

Distracters:

A & B are plausible because the feedwater flow instruments are the primary means for determining core thermal power. The applicant may recall that the RWLs limits are reactor power dependent and select the feedwater flow instrument as the method of determining reactor power for the RWM.

D is plausible, but incorrect. If turbine 1st stage pressure fails upscale, the RWL will sense a high power level above the high pressure setpoint. With reactor power at 90% it would have no effect on the operation of the RWL. If reactor power was initially low, and the turbine 1st stage pressure failed upscale, the RWL would allow less notches of rod withdrawal and be more conservative.

K/A Match

The applicant must understand the relationship between the pressure control system and Rod Control and Information System (RC&IS). The Rod Worth Limiter is a subsystem of RC&IS which provides control rod withdrawal blocks at high power.limits

Technical References:

R-STM-0500, Rod Control and Information System

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:
RLP-STM-0500-LO, OBJECTIVE 2: Describe the RC&IS automatic function and interlocks associated with the following (2)
a) Control Rod Blocks
b) Rod Pattern Controller
c) Rod Withdrawal Limiter

Question Source:	**Bank #**	

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	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(B)(6)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	RO
202001 Recirculation System Knowledge of electrical power supplies to the following: K2.02 MG sets: Plant-Specific	**Tier**	2
	Group #	2
	K/A	202001 K2.02
	Rating	3.2
	Revision	1
Revision Statement:		

Question: 29

What is the power supply to the motor generator set for Recirculation pump A?

A. NPS-SWG1A

B. NJS-SWG1L

C. NHS-MCC2A

D. NJS-SWG2A

Answer: B

Explanation:

Per figure 6 of STM-53, page 83, this is the power supply to the A recirc. Pump motor generator set.

Distracters:

A This is the power supply for fast speed operation of the A recir.pump. The unprepared applicant may confuse this with slow speed operation.(STM-53 page 4, figure 6 of STM-53, page 83)
C This also is a 480VAC power supply to other recirc. Pump A components.(STM 53 page 7) The unprepared applicant may confuse this with the power supply to the motor generator set.
D This also is a 480VAC power supply to other A components The unprepared applicant may confuse this with the power supply to the motor generator set.

K/A Match This question directly asks the power supply to the recirc. Pump motor generator set.		
Technical References: STM-53 REACTOR RECIRCULATION SYSTEM		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-053 Objective 18: Identify the power supplies to the following Reactor Recirculation System components: (18) Recirculation pump motors Low Frequency Motor Generator (LFMG) Sets		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(3)	
Level of Difficulty:	2	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
202002 Recirculation Flow Control System K3. Knowledge of the effect that a loss or malfunction of the RECIRCULATION FLOW CONTROL SYSTEM will have on following: K3.01 Core flow	Tier	2
	Group #	2
	K/A	202002 K3.01
	Rating	3.5
	Revision	2
Revision Statement: Swapped (1) and (2) in stem. Edited to "Which Technical Specification LCO is no longer satisfied?"		

Question: 30

The plant is operating at 100% reactor power.

Reactor Feed Pump 'A' trips.

The Hydraulic Power Unit for Recirc Flow Control Valve 'A' trips at the start of valve movement.

When verifying recirculation loop jet pump flow mismatch with both recirculation loops in operation, both loop flows must be within ≤___(1)___% rated core flow when operating at ≥70% of rated core flow.

(2) Which Technical Specification LCO is no longer satisfied?
3.4.1, Recirculating Loops Operating
3.4.3, Jet Pumps

	(1)	(2)
A.	5	3.4.1
B.	10	3.4.1
C.	5	3.4.3
D.	10	3.4.3

Answer: A

Explanation:
Per Technical Specifications:
LCO 3.4.1 A. Two recirculation loops shall be in operation with matched flows.
SR 3.4.1.1 Verify recirculation loop jet pump flow mismatch both recirculation loops in

operation is:

≤ 5% of rated core flow when operating at
≥ 70% of rated core flow.

When the Reactor Feed Pump trips the A and B Recirc FCVs should runback. The A failed to runback due to HPU trip. This malfunction will create a mismatch greater than 5%.

Distracters:

(1) 3.4.3 is plausible if applicant applies jet pump flow mismatch surveillance requirement to 3.4.3 instead of 3.4.1.

(2) 10% is plausible if applicant confuses requirement for <70 of rated core flow.

K/A Match

The applicant must have knowledge of the effect that the tripped HPU will have on the FCV during the required FCV runback when the feedpump tripped. The applicant must also understand the plant implications of the failure of the FCV to runback to match jet pump flow to be able to properly apply the correct technical specification requirement.

Technical References:

Technical Specification 3.4.1 and 3.4.3

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-53-LO, Objective 11: Identify the Technical Specifications, Technical Requirements Manual, and/or Bases requirements for the Reactor Recirculation System (11)

Question Source:	**Bank #**	
	Modified Bank #	
	New	X

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Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(B)(2)	
Level of Difficulty:	3	
SRO Only Justification: **30** N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
233000 Fuel Pool Cooling and Clean-up 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	Tier	2
	Group #	2
	K/A	233000 K4.06
	Rating	2.9
	Revision	1
Revision Statement:		

Question: 31

The design feature of the Fuel Pool Cooling and Cleanup System (SFC) which prevents an inadvertent draining of the pools in the event of a suction pipe break is/are __________.

A. SFC pump trip due to low flow condition

B. containment isolation due to low pool level

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

D. anti-siphon devices installed on SFC piping

Answer: D

Explanation:

“Anti-siphon devices are installed on all SFC piping to and from the pools in the event of a piping break. The anti-siphon devices prevent inadvertent draining of the pools below the minimum level required for cooling and shielding the irradiated fuel bundle stored in the pools”
STM-602 page 39

Distracters:

A While a low flow condition would occur during a pipe break event in the SFC system, the SFC pumps do not have a low flow trip.(STM-602 page 18)

B 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.(AOP-3 pages 12, 15, 16 and 21)

C 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. .(AOP-3 pages 12, 15, 16 and 21)

K/A Match

Applicant must have knowledge of the design feature for the fuel pool cooling and cleanup system that prevents inadvertent draining of the fuel pool.

Technical References:

STM-602, Fuel Pool Cooling and Cleanup System
AOP-3, Automatic Isolations

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

STM-0602 Obj 2e

Question Source:	**Bank # Mar 2014 NRC Q59**	**X**
	Modified Bank #	
	New	

Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	

10CFR Part 55 Content:	CFR 55.41(b)4	
Level of Difficulty:	3	

SRO Only Justification: NA

PRA Applicability:

Examination Outline Cross Reference	Level	RO
234000 Fuel Handling Equipment K5. Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT : K5.02 †Fuel handling equipment interlocks	Tier	2
	Group #	2
	K/A	234000 K5.02
	Rating	3.1
	Revision	2
Revision Statement: Replaced distractors B and D with plausible control rod withdrawal block distractors		

Question: 32

The reactor is shut down and refueling operations are underway at shift turnover.

The following annunciator is received at the H13-P680 panel:

CONTROL ROD WITHDRAWAL BLOCK (P680)

What is the cause of this alarm?

A. Refueling Platform has moved into a restricted zone.

B. Moving the Mode Switch from STARTUP to REFUEL with one control rod withdrawn.

C. The Refueling Platform Main Hoist is loaded and over the core.

D. With a control rod selected and all control rods inserted, the Rod Select Clear pushbutton is depressed on the Operator Control Module.

Answer: C

Explanation:

Per R-STM-0055, Refueling System, Rod Block No. 1 light provides indication that a control rod block signal has been generated by RC&IS whenever the Refueling Platform is over the core with the main hoist loaded (i.e. .550 lbs.).

Distracters:

All distractors if applicant confuses rod block with other refueling interlocks.

A. Safety Travel Interlock light provides indication from the Zone Computer that the

Refueling Platform has moved into a restricted zone. When this light is illuminated, the bridge, trolley, and main hoist are inoperative.

B. With the mode switch in STARTUP and one control rod withdrawn, a Rod withdrawal block will be present unless it is an in-sequence rod. Moving the mode switch to REFUEL will inhibit any outward rod motion but it does not cause a withdrawal block alarm.

D. With a control rod selected and all control rods inserted, when the Rod Select Clear pushbutton is depressed, the selected rod is deselected and a "Refueling Mode Select Permissive" light energizes on the back panel. No withdraw block occurs.

K/A Match

The applicant must be knowledgeable about operational implications of moving fuel handling equipment and inputs to different refueling interlocks.

Technical References:

R-STM-0055, Refueling System

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0055-LO, Objective: (5): Given any of the following Control Room panel P680 and P870 Refueling associated annunciators, state the possible causes of the alarm and status of the Refueling System. (5)
Control Rod Withdraw Block

Question Source:	**Bank # RBS-OPS-07373**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(6)	
Level of Difficulty:	3	

SRO Only Justification: N/A
PRA Applicability:

Examination Outline Cross Reference	Level	RO
239001 Main and Reheat Steam System Knowledge of the effect that a loss or malfunction of the following will have on the MAIN AND REHEAT STEAM SYSTEM: K6.10 ADS/low low set: Plant-Specific	Tier	2
	Group #	2
	K/A	239001 K6.10
	Rating	3.6
	Revision	1
Revision Statement:		

Question: 33

The plant is operating at 100% power when a malfunction in the Low-Low set logic causes all of the Low-Low set valves to open.

Steam flow to the turbine will be reduced by ___________.

A. 12-14%

B. 18-21%

C. 24-28%

D. 30-35%

Answer: D

Explanation:

RBS has five SRVs associated with the Low-Low set logic. Each of these valves will pass 6-7% steam flow. The stem indicates that all 5 valves have opened. That would equal 30-35% steam flow.(STM-109 pages9, 33 and 61.

Distracters:

A This answer is plausible if the applicant believes that there are two Low-Low set relief valves. The applicant may remember that two of the three valves on the remote shutdown panel are low low set valves ONLY and not an ADS valve.

B This answer is plausible if the applicant believes that there are three Low-Low set relief valves. The applicant may confuse with the number of SRVs operated from either remote shutdown panel.

C This answer is plausible if the applicant believes that there are four Low-Low set relief valves. There are only four Low-Low set valves that do not also have the ADS function.

K/A Match The applicant must have knowledge of the effect a failure of the Low-Low set logic (5 valves opening) would have on the Main and Reheat steam system.		
Technical References: RLP-STM-109-LO Main Steam, Rev 1		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-109-LO Objective 26: Given a set of plant parameters, predict how the Safety Relief Valve response will affect the following: (26) Steam flow indications		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	CFR 55.41(b)3	
Level of Difficulty:	3	
SRO Only Justification: **N/A**		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
241000 Reactor/Turbine Pressure Regulating System A1. Ability to predict and/or monitor changes in parameters associated with operating the REACTOR/TURBINE PRESSURE REGULATING SYSTEM controls including: A1.14 Pressure setpoint/pressure demand	Tier	2
	Group #	2
	K/A	241000 A 1.14
	Rating	3.4
	Revision	1
Revision Statement:		

Question: 34

The plant is performing a startup per GOP-1, Plant Startup, at approximately 100 psig.

Prior to resetting the condenser vacuum BPV INHIBIT, the ATC should verify Turbine Pressure Regulator Setpoint is __(1)___ current Reactor Pressure, to ensure the bypass valves remain ___(2)___.

	(1)	(2)
A.	above	open
B.	above	closed
C.	below	open
D.	below	closed

Answer: B

Explanation:

Per GOP-1, Plant Startup, resetting the condenser vacuum BPV INHIBIT on HMI screen 5532 PRESSURE CONTROL with reactor pressure greater than the Turbine Pressure Setpoint causes the bypass valves to open.

On HMI screen 5532 PRESSURE CONTROL, verify the Turbine Pressure Regulator Setpoint is above current Reactor Pressure.

Distracters:

A. Plausible if applicant understands the pressure setpoint must be above reactor pressure, but confuses that this condition opens the bypass valve. Applicant may confuse the pressure regulator setpoint with valve that opens to raise pressure.

C. Plausible if applicant believes the bypass valves need to remain open. If the pressure setpoint is below reactor pressure the bypass valve will open.

D. Plausible if applicant confuses the effect of the bypass valve. If the bypass valve opens to raise pressure and closes to reduce pressure. This misunderstanding would lead applicant to keep the bypass valves open. Compares turbine bypass valve pressure control operation to heat exchanger bypass valve temperature control operation. The heat exchanger bypass valve is open to limit cooldown.

K/A Match

Applicant must understand the relationship requirements between pressure setpoint and reactor pressure during the reactor startup to ensure proper operation.

Technical References:

Per GOP-1, Plant Startup

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0509, EHC, Objectives 7, 9
7. Describe how the following controls, indications, and/or interlocks affect the status of the Turbine Electro-Hydraulic Control System:
    a) Pressure Set
9. Describe the Turbine EHC response and parameters to monitor from the Main Control Room during the following:
Turbine roll
Reactor pressure changes
Pressure setpoint changes

Question Source:	**Bank #**	
	Modified Bank #	
	New	X

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
245000 Main Turbine Generator and Auxiliary Systems Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.05 Generator trip	Tier	2
	Group #	2
	K/A	245000 A2.05
	Rating	3.6
	Revision	2
Revision Statement: Modified stem to, "… the main generator motoring time limit is __(1)__ …"		

Question: 35

The main generator has exceeded a trip set point. The ATC operator observes the main turbine has tripped, but the main generator output breakers have not opened.

Main condenser vacuum is 28.9"

Per AOP-2, Main Turbine and Generator Trips, the main generator motoring time limit is __(1)__ and the ATC operator should manually __(2)__ to cause the generator output breakers to open.

	(1)	(2)
A.	20 minutes	adjust VARs to 0
B.	20 minutes	trip the exciter field breaker
C.	90 seconds	adjust VARs to 0
D.	90 seconds	trip the exciter field breaker

Answer: A

Explanation:

Guidance for this condition is given in AOP-2 Main Turbine and Generator Trips. The table on page 6 gives the conditions for generator motoring time limits. For the conditions given in the stem the limit is 20 minutes. The corrective action for the given conditions is on page 7. These actions will force the generator directional relay to

operate. (SOP-80 page 83)
Distracters: B Part 1 is correct. Part 2 is an action that is taken after the generator output breakers are open in the AOP however the action is not allowed per the Caution on page 9. C Part 1 is the incorrect time per the table on page 6 of the AOP. The unprepared applicant may not remember the requirements of the table. Part 2 is correct. D Part 1 is the incorrect time per the table on page 6 of the AOP. The unprepared applicant may not remember the requirements of the table. Part 2 is an action that is taken after the generator output breakers are open in the AOP however the action is not allowed per the Caution on page 9.
K/A Match The applicant must understand the operational impact that a generator trip with a failure of the output breakers to open will have and follow AOP-2 procedural guidance to correct the condition based on the given condenser vacuum.
Technical References: AOP-2 Main Turbine and Generator Trips SOP-80 TURBINE GENERATOR OPERATION
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: RLP-HLO-521 Objective 9: Describe the Immediate (if applicable) and Subsequent Operator Actions of AOP-0002, Main Turbine Trip / Generator Trip.

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X

	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)4&10	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Risk Events: Reactor Trip/Turbine Trip		

Examination Outline Cross Reference	Level	RO
256000 Reactor Condensate System A3. Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including: A3.05 Lights and alarms	Tier	2
	Group #	2
	K/A	256000 A3.05
	Rating	3.0
	Revision	2
Revision Statement: Distractor C, capitalized AND		

Question: 36

Based on the valve pictured below, when will both lights automatically energize?

This image cannot currently be displayed.

A. One condensate pump breaker closed and low flow condition.

B. One condensate pump breaker closed and high discharge header pressure

C. Low flow condition AND high discharge header pressure

D. Low flow condition ONLY.

Answer: A

Explanation:

The photo indicates 1CNM-FV114, Short Cycle Recirc, is closed.

Per R-STM-0104, Condensate, The condensate recirculation valve modulates to maintain Condensate System flow within a specified operating band. Condensate Recirculation Valve automatically opens: When at least one Condensate Pump is running (CNM-SOV114 de-energized) and is proportionately modulated open when condensate flow starts to lower below the controller setpoint.

Distracters:

All distractor combinations are plausible if student confuses condensate minimum flow valve with RCIC Minimum Flow to Suppression Pool Valve.

Per STM-0209, RCIC, Following the cooling water header tap-off of the pump discharge header, another line tap is provided to ensure minimum flow is maintained through the pump. Flow through the minimum flow line is controlled by E51-F019, RCIC Minimum Flow to Suppression Pool Valve. Although normally closed, F019 will open upon a low flow 120 gpm condition as sensed by a flow element in the RCIC pump discharge header, and header pressure >125 psig.

K/A Match

Applicant must first understand given light indications for the condensate system short cycle recirc valve in photo. Then apply current status of valve position to what both lights energizing means and what conditions make that happen. The applicant must understand the valve is currently closed and what are the requirements to open the recirc valve.

Technical References:

R-STM-0104, Condensate System
R-STM-0209, RCIC System

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0104,-LO, Condensate, Objective (5): Describe the following controls, indications and/or interlocks/automatic function for the following: (5)
Minimum Flow Recirculation Valve, CNM-FV114

Question Source:	**Bank #**	
	Modified Bank #	
	New	X

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

2018 RBS NRC Examination

Examination Outline Cross Reference	**Level**	RO
259001 Reactor Feedwater System Ability to manually operate and/or monitor in the control room: A4.08 FWRV position	**Tier**	2
	Group #	2
	K/A	259001 A4.08
	Rating	3.3
	Revision	2
Revision Statement: Modified stem: “What is current RPV level setpoint, approximately?”		

Question: 37

Based on the indications below:

What is current RPV level setpoint, approximately?

A. 35 inches

B. 50 inches

C. 75 inches

D. 80 inches

Answer:	**A**

Explanation:

Per RLP-STM-107-FWLC-LO, Rev 1, the FW REG Valves Master Flow Controller in in automatic (Green light ON) set at 35 inches (red line next to thumb wheel).

Distracters:

B. FWREG Valve A and C Flow Controllers are both set at approximately 50 inches. Plausible if applicant believes each flow controller controls the FWRV position separately.

C. FWREG Valve A Output demand is approximately 75. Plausible if applicant does not understand the flow controller indications.

D. FWREG Valve B and C Flow Controllers output demand are at approximately 80.

K/A Match

Applicant must be able to monitor proper FWRV operation based on the master flow controller setpoint and demand signal.

Technical References:

RLP-STM-107-FWLC-LO,Feedwater Level Control System

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:
RLP-STM-0107-FWS-LO Rev 0, Objective 5
Describe the purpose and operation of the following Feedwater Level Control System components: (5)

- Startup Reg Valve Controller
- FW Reg Valve Controller
- FW Reg Valves
- FW Reg Master Controller

Question Source:	**Bank #**	
	Modified Bank #	

	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)4	
Level of Difficulty:	2	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
288000 Plant Ventilation Systems 2.4.21 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.	Tier	2
	Group #	2
	K/A	288000 2.4.21
	Rating	4.0
	Revision	2
Revision Statement: Provided additional information in the K/A match section of the question of the relationship between Plant Ventilation Systems K/A, Standby Gas Treatment System, Reactor Building Annulus Ventilation, and the Radioactive Release Safety Function.		

Question: 38

Which of the following signals cause automatic starting of both trains of Standby Gas Treatment?

A. RPV Level 3

B. Containment Purge Isolation Radiation - High

C. Fuel Building Ventilation Exhaust Radiation – High

D. Reactor Building Annulus Ventilation Gaseous Radiation - Hi-Hi

Answer: D

Explanation:

Per R-STM-0257, Standby Gas Treatment, Both trains of standby gas treatment will auto start if the following are present:
- Inlet damper GTS-AOD1A(B) fully open
- 480 VAC bus voltage available;
- 4160/480 VAC standby local sequencing permissive satisfied; and
- Inverse time and instantaneous overcurrent trip reset
- Pushbutton for SGTS exhaust fan in "AFTER STOP" condition and Lockout/Reset pushbutton in "RESET", and any of the following:
- LOCA signal (1 out of 2, twice) Level 2(-43"), 1.68 psid Drywell pressure
- Low annulus pressure control system flow (180 cfm)
- RMS RE11A(B) Reactor building annulus ventilation gaseous radiation hi-hi (3.89 x 10-5 Tci/cc).

Distracters:

A. Plausible if applicant confuses which reactor level signal starts both standby gas treatment fans.
B. Plausible if applicant confuses with Group 8 valve isolation signal. Containment Purge Isolation Radiation - High: 1.17 R/hr; Group 8 valves isolate.
C. Plausible if applicant confuses with Group 13 damper isolation signal. Fuel Bldg Vent Exhaust Rad - High: 1.64 x 103 mCi/sec, RMS-RE5A 5.29 x 10-4 mCi/cc, RMS-RE5B; Group 13 dampers isolate.

K/A Match

The NUREG 1123 safety function for Plant Ventilation Systems is Radioactivity Release. Therefore, the components of the KA that this question was designed to match are:

2.4.21 Knowledge of …parameters and logic used to assess the status of safety function(s)…(for) Radioactivity Release.

The Plant Ventilation Systems, specifically the Reactor Building Annulus Ventilation System, includes sensors and logic to detect a radioactive release.

When a radioactive release is detected by Reactor Building Annulus Ventilation System, initiation signals realign annulus ventilation dampers (Group 18 Isolation) to isolate normal annulus ventilation and initiate Standby Gas Treatment process flow low to mitigate the radioactive release. (i.e secondary containment isolation)

Therefore, to assess the status of the Radioactive Release safety function, an applicant needs to be knowledgeable of the parameters and logic that detect a radioactive release and cause automatic realignment of annulus ventilation dampers (Group 18 isolation) and initiate the Standby Gas Treatment.

The question tests the applicant’s knowledge with regards to the status of Reactor Building Annulus Ventilation Gaseous Radiation - Hi-Hi condition. Of all the signals that isolate the reactor building annulus and initiates Standby Gas Treatment, this is the only signal detects and terminates a radioactive release in progress which is a direct tie to assess the Radioactive Release safety function. The other initiation signals are “anticipatory” of a radioactive release (LOCA signal, low reactor water level, or low annulus pressure control system flow).

Additional supporting information is demonstrated by the interrelationship between Plant Ventilation Systems and Standby Gas Treatment System from the following KA from NUREG 1123:

Knowledge of PLANT VENTILATION SYSTEMS design feature(s) and/or interlocks which provide for the following:
K4.01 Automatic initiation of standby gas treatment system 3.7

K4.02 Secondary containment isolation 3.7		
Technical References: R-STM-0257, Standby Gas Treatment AOP-3, Automatic Isolations		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0257-LO, Standby Gas Treatment, Objective 5: Discuss the operation of the Standby Gas Treatment System including: (5) a) Automatic initiation/trip signals		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(9)	
Level of Difficulty:	2	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295001 Partial or Complete Loss of Forced Core Flow Circulation Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : AA2.02 Neutron monitoring	Tier	1
	Group #	1
	K/A	295001 / AA2.02
	Rating	3.1
	Revision	1
Revision Statement:		

Question: 39

Following a trip of Recirc Pump ‘A’ from 75% power, operators are inserting control rods to exit the Restricted Region.

While in the Restricted Region, which of the following would require an immediate manual scram per AOP-0024, THERMAL HYDRAULIC STABILITY CONTROLS?

A. Fraction of Core Boiling Boundary (FCBB) > 1.0 while in the Restricted Region.

B. Entry into the Restricted Region AND a Loss of Feedwater Heating event occurs.

C. Unexpected entry into the Restricted Region occurs AND one channel of PBDS is inoperable.

D. Valid alarm P680-07A-A06 (B06), DIV 1(2) PERIOD BASED DETECTION SYSTEM HI-HI DECAY RATIO.

Answer: D

Explanation:

This is the correct answer per step 5.3.2 of AOP-24 page 8. For this to be a valid alarm the operator must determine that both APRM A/B PBDS Highest confirmed count and APRM A/B PBDS 2nd Highest Confirmed Count is greater than 11 Successive Confirmation Counts (SCCs). AOP-24 step 5.3 page 8.

Per P680-07A-A06 (B06), DIV 1(2) PERIOD BASED DETECTION SYSTEM HI-HI DECAY RATIO, the initiating device for this alarm is APRM A PBDS Card Successive Confirmation Counts (SSC) 1. At least 2 LPRMs with greater than or equal to 11 SCCs.

Distracters:

A The Fraction of Core Boiling Boundary (FCBB). If the FCBB is greater than one boiling is occurring lower in the core which could aggravate any power oscillations. There are no requirements to scram the reactor for this condition. (AOP-24 step 5.4 page 8 and Tech Spec 3.2.4)

B A loss of feed heating event would cause reactor power to rise while in the low flow condition of the restricted region and could aggravate any power oscillations. There are no requirements to scram due to these conditions.(AOP-24 step 5.5 page 8)

C This would be correct if the question stem indicated that BOTH channels of PBDS were inoperable. The question does not indicate that any of the nuclear instruments are inoperable. (AOP-24 step 4.3, page 6)

K/A Match

The question requires the applicant to interpret the indications from neutron monitoring instrument during a partial loss of forced circulation based on the valid alarm.

Technical References:

AOP-24 THERMAL HYDRAULIC STABILITY CONTROLS
TS 3.2.4

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-OPS-AOP24, Objective 5: Describe the immediate (if applicable) and subsequent operator actions associated with AOP-0024, Thermal Hydraulic Stability Controls. (5)

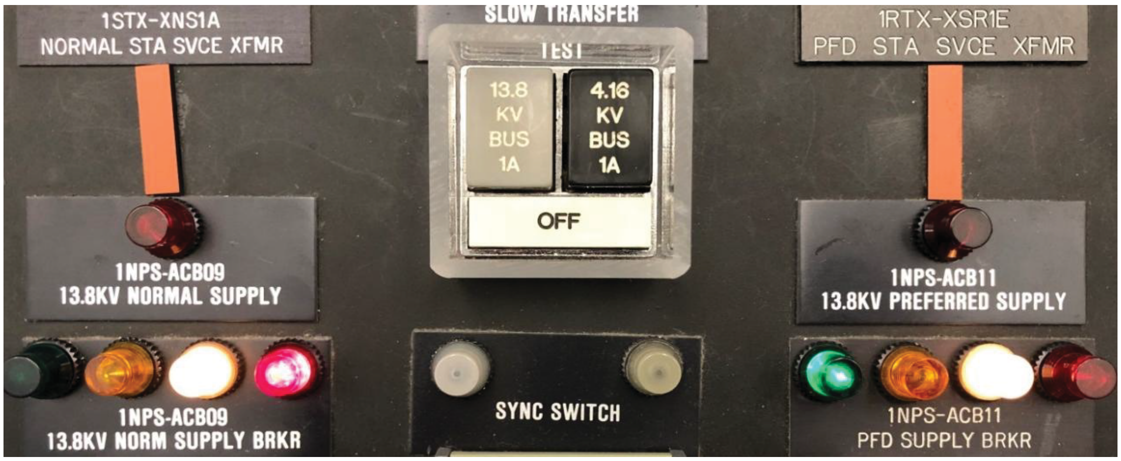
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(10)	

Level of Difficulty:	3	
SRO Only Justification: **N/A**		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	RO
295003 Partial or Complete Loss of A.C. Power 2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	**Tier**	1
	Group #	1
	K/A	295003 2.1.31
	Rating	4.6
	Revision	2
Revision Statement: Corrected label nomenclature in the stem 1RTX-XSR1D to 1RTX-XSR1F		

Question: 40

Consider the following plant conditions (1STX-XNS1A, 1RTX-XSR1E, 1STX-XNS1C, and 1STX-XNS1C Pictured, STX-XNS1B & 1RTX-XSR1F (not pictured) are aligned the same):



Following a turbine trip and a slow transfer, which of the following would be expected as a result?

A. HVN-P1A, VENT CHILL WTR trips.

B. CCS-P1B, CLG WTR PUMP MOT, auto starts due to a trip of the running pump.

C. CNM-P1A, CNDS PUMP 1A trips.

D. All three diesel generators auto start.

Answer: **C**

Explanation:

Per R-STM-0300, AC Distribution, for the slow transfer mode, bus voltage is allowed to drop to 25% of rated to ensure all loads have dropped off prior to restoring power.

CNM-P1A is supplied by NPS-SWG1A which was supplied by Normal transformers prior to the transient so would be affect by the slow transfer.

Distracters:

All distractors are plausible if applicant confuses power supply or plant lineup.

A. Turbine building chilled water pumps are supplied by 4160VAC NNS-SWGR1A,B,C which are fed from preferred power so would not be affected by a slow transfer.

B. CCS-P1A & C are supplied by 4160VAC NNS-SWGR1A & C which are fed from preferred power so would not be affected by a slow transfer.

D. DGs are supplied by NNS which are supplied by preferred transformers, so would not be affected by a slow transfer.

K/A Match

Applicant must be able to interpret control room indications to correctly identify the result of transient.

Technical References:

R-STM-0300, AC Distribution, Revision 31
EE-001AC, Rev 56

Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0300 Obj. 3: Describe the function and operation of the following AC Distribution System components: (3) Slow Transfer Test		
Question Source:	**Bank # Nov 2010 Audit**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(4)	
Level of Difficulty:	4	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Risk Significant Systems: NPS 13.2 kV AC Power		

Examination Outline Cross Reference	Level	RO
295004 Partial or Complete Loss of D.C. Power Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : AK1.06 Prevention of inadvertent system(s) actuation upon restoration of D.C. power	Tier	1
	Group #	1
	K/A	295004 / AK1.06
	Rating	3.3
	Revision	3
Revision Statement: Revised question to test knowledge of step 5.9 of AOP 14 to verify that all loads fed from the bus are deenergized and isolated. Distractors are different combinations of load types.		

Question: 41

Following a loss of DC bus BYS-PNL03B, maintenance reports that BYS-PNL03B can be restored.

In accordance with AOP-14 LOSS OF 125VDC, which loads must be deenergized / isolated prior to reenergizing DC bus BYS-PNL03B?

A. Pump motors ONLY

B. Pump motors and Automatic Containment Isolation Valves ONLY

C. Automatic Containment Isolation Valves and actuation/initiation logic circuits ONLY

D. All loads

Answer: D

Explanation:

DC bus BYS-PNL03B supplies pumps, motor operated valves, and logic circuits. Step 5.9 of AOP-14 states: “Prior to re-energizing any bus, verify that all loads fed from the bus are de-energized and isolated.”

Distracters:

A Pump motors only is plausible because the concern for surge current when

reenergizing the bus for auto starting motors. B Pump motors is plausible as given in A. Automatic containment isolation valves is plausible due to unanticipated containment isolations occurring when the bus is re-energized. C Automatic containment isolation valves are plausible as given in B. Actuation / initiation logic circuits is plausible due to unanticipated initiations or actuations of logic upon reenergizing the bus.
K/A Match This question tests the knowledge of the applicant concerning the procedural requirement for deenergizing/isolating loads prior to restoration of DC power.
Technical References: AOP-14 LOSS OF 125VDC
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: RLP-OPS-AOP14, Objective 10: Describe the immediate (if applicable) and subsequent operator actions associated with AOP-0014, Loss of 125VDC (10)

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	CFR: 41(b)(8)&(10)	
Level of Difficulty:	3	
SRO Only Justification: NA		

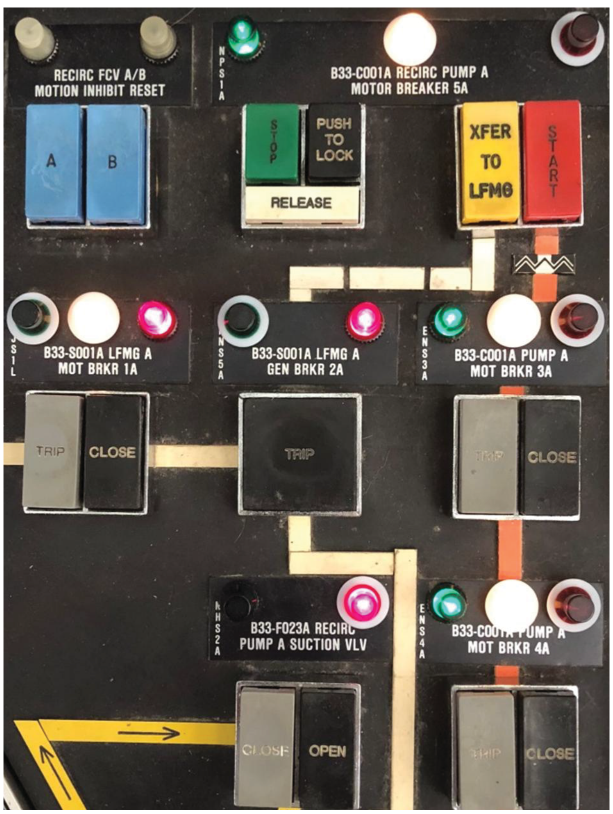
PRA Applicability:
Top 10 Risk Significant Systems: Safety Related DC Power

Examination Outline Cross Reference	Level	RO
295005 Main Turbine Generator Trip AK2. Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: AK2.03 Recirculation system	Tier	1
	Group #	1
	K/A	295005 AK2.03
	Rating	3.2
	Revision	2
Revision Statement: Revised distractor C to “manually shifted” for consistency with D		

Question: 42

Plant was operating at 100% power.

Based on indications below, what transient occurred?



A. Main Turbine Tripped

B. Low reactor water level transient.

C. Recirc Pumps were manually shifted from slow to fast speed.

D. Recirc Pumps were manually shifted from fast to slow speed.

Answer: A
Explanation: The "End-of-Cycle" Recirc Pump Trip (EOC-RPT) interlock initiates an automatic transfer to low speed whenever a turbine stop valve closure or control valve fast closure occurs and power is greater than 30.4% (as sensed by first stage turbine pressure). CB-3 and CB-4 each have two trip coils. Trip coil number 1 is utilized for the EOC-RPT logic.
Distracters: All distractors are plausible if applicant cannot correctly interpret plant conditions given in picture. B. Plausible if applicant confuses EOC-RPT trip with ATWS RPT trip. The ATWS recirculation pump trip (RPT) actuates to trip the recirculation pumps to OFF as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. Tripping the recirculation pumps to OFF greatly increases the void content in the core, adding a large amount of negative reactivity. The trip signals to actuate the ATWS interlock are Reactor Vessel Low Level, Level 2 (-43 inches) or Reactor High Pressure (1153 psig). This interlock opens CB-2 (LFMG Output) breaker, CB-5 (fast speed supply) breaker, and initiates ARI (Alternate Rod Insertion) whenever the ATWS logic circuitry is satisfied. The opening of CB-5 causes CB-3 to open. The CB-4 would remain closed. C. When CB-5 START is depressed the LFMG breakers CB-1 and CB-2 trip and the pump is allowed to slow down to less than 20% speed, before all permissives to close CB-5 are met. (Refer to Sect 1.4.1.9 for permissives) When CB-1 and CB-2 trip, a 3 second timer starts, preventing closure of CB-5. This allows for pump coastdown. After the time delay has expired, CB-5 closes when the following permissives are met: · Pump speed less than 20% (speed sensors) · CB-5 "PUSH TO LOCK" or "STOP" pushbutton NOT DEPRESSED · Pump motor lockout relay (86) reset (local) · LFMG generator lock out relay (86) reset (local) · CB-2 open (breaker auxiliary contact) · CB-4 closed (breaker auxiliary contact) D. The pump motor may be manually transferred from high speed to low speed when total feed flow is greater than 19.9%, if both "TRANSFER TO LFMG" pushbuttons are DEPRESSED on H13-PNLP680. The flow control valve controllers automatically shift to manual. With both pushbuttons DEPRESSED, CB-5 trips immediately. CB-3 will trip open and CB-4 will remain closed.

K/A Match Applicant must have knowledge to interpret recirc pump plant conditions and assess the interrelationship with a main turbine trip.		
Technical References: R-STM-0053, Reactor Recirculation System, Revision 15		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0053-LO, Reactor Recirc System Rev 2, Objective (2): Describe the following automatic features and interlocks for the Reactor Recirculation System components, including setpoints, system response and indications (2) End of Cycle Recirculation Pump Trip (EOC-RPT)		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(6)	
Level of Difficulty:	4	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295006 SCRAM Knowledge of the reasons for the following responses as they apply to SCRAM : AK3.04 Reactor water level setpoint setdown: Plant-Specific..	Tier	1
	Group #	1
	K/A	295006 / AK3.04
	Rating	3.1
	Revision	1
Revision Statement:		

Question: 43

Five seconds after the initiation of the Feedwater Level Control Setpoint Setdown feature, the reactor water level control setpoint is set at ___(1)___ in order to control the feedwater flow transient to limit the resulting ___(2)___ following a high power scram.

	(1)	(2)
A.	18 inches	high reactor water level
B.	9.7 inches	high reactor water level
C.	18 inches	low reactor water level
D.	9.7 inches	low reactor water level

Answer: A

Explanation:
In the event reactor water level decreases to Level 3 with the Master Flow Controller in automatic, Setpoint Setdown is initiated. When initiated, it causes the Master Flow Controller to change the level setpoint to 18” after a 5 second time delay. The purpose of Setpoint Setdown is to prevent a high reactor level (Level 8) him trip of the feed pumps following a scram. (STM-107 page 63)

Distracters:
B Setpoint is incorrect, but plausible. 9.7” is the initiation setpoint for Setpoint Setdown.
C Setpoint is correct. Reason is incorrect but plausible. The reason for setpoint set down is to prevent a high reactor level condition following a high power scram; however, 18 inches was chosen as the setpoint (as opposed to a lower setpoint) to prevent reducing feed flow too soon and causing excessively low reactor level.
D Setpoint is incorrect, but plausible as given in B.. Reason is incorrect but plausible as given in C.

K/A Match

The applicant must understand reactor water level setpoint setdown response after a reactor scram and the reason for the response..		
Technical References: STM-107 REACTOR FEEDWATER AND LEVEL CONTROL SYSTEMS		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-107-FWLC-LO Objective 4: State the purpose, setpoints & operation of the Setpoint Setdown feature of the Master Flow Controller (4)		
Question Source:	**Bank #**	2010 Audit exam
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b(5)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295016 Control Room Abandonment AA1. Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT : AA1.08 Reactor pressure	Tier	1
	Group #	1
	K/A	295016 AA1.08
	Rating	4.0
	Revision	1
Revision Statement:		

Question: 44

The Main Control Room has been abandoned due to a fire.

Transfer to the Remote Shutdown panel has been completed including, the de-energization of ENB-PNL02A and ENB-PNL02B in accordance with AOP-31, Shutdown From Outside the Main Control Room.

On C61-P001, the LOCAL SVV EMERGENCY CONTROL ALIGNED light is on.

___(1)___ SRVs will be available for the ___(2)___ modes at the Remote Shutdown Panels.

	(1)	(2)
A.	3	manual, ADS (1 only), safety, Low Low Set, and relief
B.	3	safety and manual ONLY
C.	13	manual, ADS (1 only), safety, Low Low Set, and relief
D.	13	safety and manual ONLY

Answer: B

Explanation:

Complete transfer to the Remote Shutdown System provides for operation of 3 SRVs from either RSS panel in safety and manual modes only. The de-energization of the ENB panels prevents electronic operation of the 13 remaining SRVs therefore they are only available in the Safety mode.

Per R-STM-0200, Remote Shutdown System, three (3) Safety Relief Valves (SRVs), B21-F05lC, D and G can be operated from either RSS Room. All three normally operate in three modes, safety, LLS, and relief, with B21-F051G also capable of operating as an Automatic Depressurization System (ADS) valve. When the associated

RSTS is in EMERGENCY, the relief and LLS mode of B21-F051C, D and G, and the ADS mode of B21-F051G are rendered inoperative. All three valves are still able to operate in the safety mode or can be opened manually from either RSS Station.

Distracters:

A. Three is correct number of SRVs available. Plausible if applicant does not remember what happens when the LOCAL SVV EMERGENCY CONTROL ALIGNED light is on. This light indicates the relief, LLS, and ADS modes are no longer available. Prior to transferring RSTS switch to emergency, these modes were available to the three SRVs at the remote shutdown panel.

C. 13 is plausible number of SRVs if applicant confuses with number of SRVs remaining in safety mode only. Plausible if applicant does not remember what happens when the LOCAL SVV EMERGENCY CONTROL ALIGNED light is on. This light indicates the relief, LLS, and ADS modes are no longer available. Prior to transferring RSTS switch to emergency, these modes were available to the three SRVs at the remote shutdown panel.

D. 13 is plausible number of SRVs if applicant confuses with number of SRVs remaining in safety mode only. Safety and manual modes are the only modes available.

K/A Match

Applicant must understand which SRVs are available to monitor and manually operate to control reactor pressure at the remote shutdown panel during control room abandonment due to a fire.

Technical References:

AOP-31, Shutdown from Outside the Main Control Room
R-STM-200, Remote Shutdown System
R-STM-109, Main Steam System

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-OPS-AOP31, Objective 4: Describe the immediate (if applicable) and subsequent

operator actions associated with AOP-0031, Shutdown from Outside the Main Control Room. (4)		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295018 Partial or Complete Loss of Component Cooling Water 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	Tier	1
	Group #	1
	K/A	295018 / AA2.03
	Rating	3.2
	Revision	1
Revision Statement:		

Question: 45

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 annunciators is______________.

A. High drywell pressure

B. Low reactor water level

C. 50 psig CCP header pressure

D. Manual initiation of Standby Service Water

Answer: A

Explanation:

High drywell pressure isolates CCP to containment resulting in loss of cooling to the RWCU heat exchanger and Recirc Pump seal and winding coolers. This causes all of the listed alarms to illuminate.(STM-115 page16, 38 and AOP-3 page 12, 16)

Distracters:
B The CCP isolation valves would not close until reactor water level falls to level 2 (-43”). (AOP-3 page 12, 16) the unprepared applicant may confuse the isolation signal setpoint.
C With pressure this low, the CRD pumps would trip and the indicated charging water header pressure would be 0 psig.(STM-115 page 13) The unprepared applicant may not recognize the correlation between low CCP pressure and the charging water pressure.
D Manual initiation of SSW would result in a trip of the CRD pump therefore charging water header pressure would be 0 psig. (STM-115 page 20) The unprepared applicant may not recognize the correlation between low CCP pressure and the charging water pressure.

K/A Match
The question indicates that a partial loss of CCW to equipment inside of the containment / drywell and the applicant must determine the cause of the partial loss.

Technical References:
AOP-3 pages 12, 16, STM-115 pages 13, 16, 38

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:
RLP-OPS-AOP11 Rev 3 Obj 2: Identify the symptoms of a loss of CCP.

Question Source:	**Bank # March 2014 NRC Q7**	X
	Modified Bank #	
	New	

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X

10CFR Part 55 Content:	41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295019 Partial or Complete Loss of Instrument Air 2.4.31 Knowledge of annunciator alarms, indications, or response procedures.	Tier	1
	Group #	1
	K/A	295019 2.4.31
	Rating	4.2
	Revision	4

Revision Statement:
Revision 2: Added to the explanation how the question supports the K/A and justification in this case for use of a negatively phrased question
Revision 3: Rephrased question to eliminate negatively phrased question to meet requirements of the NUREG.
Revision 4: Changed B to “start drifting closed.”

Question: 46

The following valid alarm is received:

INSTRUMENT AIR HEADER PRESSURE LOW (H13-P870/51A/B02)

An operator observes instrument air pressure is slightly below the alarm setpoint and slowly lowering.

Which of the following plant responses would be expected for current conditions?

A. SAS-AOV134, IAS-SAS CROSS TIE VLV opened

B. Main Steam Isolation Valves (MSIVs) start drifting closed

C. C33-LVF001A/B/C, FWREG VALVES A/B/C locked up

D. Control rods individually scrammed as the scram valves opened

Answer: A

Explanation:

Per AOP-8, Loss of Instrument Air, 3.1 At Instrument Air pressure of 113 psig the following occurs:
3.1.1. SAS-AOV134, IAS-SAS CROSS TIE VLV opens to cross tie the SAS air compressors to Instrument air

Per H13-P870/51A/B02, INSTRUMENT AIR HEADER PRESSURE LOW,

INITIATING DEVICES	SETPOINTS
1. IAS-PS101 1.	110 psig

OPERATOR ACTIONS:
2. Verify the Service Air and Instrument Air systems have realigned as follows:
· SAS-AOV134, IAS-SAS CROSS TIE VLV opens (IAS Air Pressure 113 psig).

Distracters:

B. Per AOP-8, Loss of Instrument Air, If Air header pressure lowers to 50 psig, then verify the MSIVs are closed.

C. Per AOP-8, Loss of Instrument Air, the Feedwater Reg Valves lock up on low air pressure less than or equal to 50 psig. The condition is instrument air pressure slightly below the alarm setpoint. Per H13-P870/51A/B02, the alarm setpoint is 110 psig. With instrument air pressure slightly less than 110 psig it would be NOT be expected that the Feed Reg Valves would be locked up.

D. Per AOP-8, Loss of Instrument Air, if any of the following occurs, then scram the reactor: instrument air header pressure lowers to 65 psig. The manual scram is required prior to individual rod scrams.

K/A Match

Applicant must have knowledge of the alarm response procedure and associated automatic actions for a loss of instrument air header pressure.

Technical References:

H13-P870/51A/B02, INSTRUMENT AIR HEADER PRESSURE LOW
AOP-8, Loss of Instrument Air

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0121-LO, Objective (3): List the automatic functions and interlocks of the following plant air system components: (3)
Air Compressors.
IAS - SAS Cross-Tie Valve (1SAS-AOV134).
Service Air System (SAS) Header Block Valve (1SAS-AOV133).
Air Dryers including IAS-AOV300A(B).

Containment Isolation Valves.		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		
Top 10 Risk Significant Systems: Instrument and Service Air		

Examination Outline Cross Reference	Level	RO
295021 Loss of Shutdown Cooling Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : AK1.01 Decay heat	Tier	1
	Group #	1
	K/A	295021 / AK1.01
	Rating	3.6
	Revision	2
Revision Statement: Edited the format of the stem. List the 4 options as 1, 2, 3, 4. Made answer choices to support the change in the format of the stem.		

Question: 47

The plant is Shutdown in Mode 4 and the following plant conditions exist:

- Main Condensers are drained and open for maintenance.
- Service Water temperature is 83°F.
- RPCCW temperature is 91°F.
- HPCS, RHR “B”, and RHR “C” pump motors are tagged out for repairs.
- Reactor Recirculation Pump “B” is tagged out for seal replacement.
- Reactor Recirculation Pump “A” is operating in Slow Speed.

RHR Loop “A” was operating in shutdown cooling mode when the RHR A pump tripped.

Incore Fuels Group has calculated decay heat as 22.5 E6 Btu/hr with a current reactor coolant temperature of 130°F

Which of the following will meet the Alternate Shutdown Cooling Methods criteria to restore and maintain reactor coolant temperature less than or equal to 130°F?

Alternate Shutdown Cooling Methods

1. SPC/ADHR
2. RWCU
3. SFC
4. CRD

A. 1 ONLY

B. 1 OR 2 ONLY

C. 1, 2 OR 3 ONLY

D. 1, 2, 3 OR 4 ONLY

Answer: A
Explanation: With the given plant conditions ADHR is the only system or combination of systems that are available for use with enough heat removal capacity to remove 22.5 E6 Btu/hr to maintain 130°F reactor water temperature. The main steam system is not available for use because the main condensers are drained and open for maintenance and MSL Flooding requires the main condenser.(OSP-41 ALTERNATE DECAY HEAT REMOVAL pages 55-60 as directed by AOP-51 LOSS OF DECAY HEAT REMOVAL page 11) Did not spell out the Alternate Shutdown Cooling Methods since OSP-41 ALTERNATE DECAY HEAT REMOVAL pages 55-60 is provided to the applicants as a handout.
Distracters: B RWCU and SFC can remove a total of 21.96 E6 Btu/hr. this is less than the 22.5 E6 Btu/hr at 130°F reactor water temperature required to be removed. C CRD and SFC can remove a total of 20.61 E6 Btu/hr. this is less than the 22.5 E6 Btu/hr at 130°F reactor water temperature required to be removed. D CRD and RWCU can remove a total of 6.35 E6 Btu/hr. this is less than the 22.5 E6 Btu/hr at 130°F reactor water temperature required to be removed.
K/A Match The question requires the applicant have knowledge of the systems to be operated to remove decay heat during a loss of shutdown cooling.
Technical References: AOP-51 page 11, OSP-41 pages 55-60
Handouts to be provided to the Applicants during exam: OSP-41 pages 55-60
Learning Objective: RLP-HLO-543 Objective 6: Given a copy of the applicable Attachments to OSP-0037 and OSP-0041, and plant conditions, determine Time to 200 deg.F, Time to Top of Active Fuel, Decay Heat Load, and alternate systems sufficient for the removal of Decay Heat. (6)

Question Source:	**Bank # 2003 NRC Q73**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(8) & (10)	
Level of Difficulty:	4	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295023 Refueling Accidents AK2. Knowledge of the interrelations between REFUELING ACCIDENTS and the following: AK2.05 Secondary containment ventilation	Tier	1
	Group #	1
	K/A	295023 AK2.05
	Rating	3.5
	Revision	1
Revision Statement:		

Question: 48

Refueling operations are in progress when an irradiated fuel bundle is dropped in the spent fuel pool. The refuel handling team on the lower bridge reports bubbles rising from the dropped fuel bundle and the following annunciators are received in the MCR:

- H13-P863-75A-H01, DIV I Fuel Bldg Exh PAM Gaseous Radn Alarm
- RMS-DSPL230-1GE005, Fuel Build Stack/Vent Exhaust A – High
- RMS-DSPL230-2GE005, Fuel Build Stack/Vent Exhaust A – High

Which of the following describes the Fuel Building ventilation lineup after the conditions given above?

A. Fuel Building Ventilation is completely isolated

B. Supply air is via normal supply fans and exhaust is through the Div 1 charcoal filter trains ONLY

C. Supply air is via normal supply fans and exhaust is through the Div 2 charcoal filter trains ONLY

D. Supply air is via Fuel Receiving Area and exhaust is through both Div 1 and Div 2 charcoal filter trains

Answer: D
Explanation: RMS-RE5A drives all three annunciators; this instrument will start the Div 1 filter train only, however a low flow signal generated from the tripping of the Fuel Building Supply Fans will also start the Div 2 train.(See page 19 of STM reference) P&L 2.1 states that 1 train will need to be secured after an auto start signal is received.

Distracters:

A. A high radiation condition in the fuel building does not isolate the ventilation system; it isolates the normal supply and exhaust fans, and starts the charcoal filtration system.

B & C Plausible if applicant confuses DIV I Fuel Bldg Exh PAM Gaseous RADN Alarm.. A high radiation condition in the fuel building isolates the normal supply and exhaust fans, and starts the charcoal filtration system.

K/A Match

Applicant must have knowledge of Fuel Building ventilation response to a high radiation condition due to a refueling accident.

Technical References:

R-STM-0406, pp 19 & 41 of 50 ;
ARP-P863-75A-H01;
SOP-0062, FB HVAC, P&L 2.1

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0406, Obj 11: Given plant/system status and key parameters, predict/determine Fuel Building HVAC System response. (11)

Question Source:	**Bank # Dec 2014 NRC Q9**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(7)&(13)	
Level of Difficulty:	4	

SRO Only Justification: N/A
PRA Applicability:

Examination Outline Cross Reference	Level	RO
295024 High Drywell Pressure Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : EK3.03 Containment venting: Mark-III	Tier	1
	Group #	1
	K/A	295024 / EK3.03
	Rating	3.6
	Revision	1
Revision Statement:		

Question: 49

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

In accordance with EOP-0002, PRIMARY CONTAINMENT CONTROL, containment emergency venting is required BEFORE containment pressure reaches 30 psig with allowance to exceed release rate limits due to ______.

A. exceeding the Pressure Suppression Pressure(PSP)

B. exceeding the code allowable stresses for SRV tailpipes

C. the inability to open ventilation dampers above this pressure

D. exceeding design inlet pressure for Standby Gas Filter Train

Answer: C
Explanation: 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.(EOP-2 Primary Containment Control step CP-8, EPSTG*0002 page B-8-20
Distracters: A Exceeding PSP limit in EOP-2 requires the operator to emergency depressurize the reactor. The unprepared applicant may confuse the bases for ED with emergency venting. B The SRV tailpipe stress limit is based on high suppression pool level of 21’3”. The

unprepared applicant may confuse the bases for SRV tail pipe limit with emergency venting. D The containment pressure leg of EOP-0002 has guidance to avoid damage to the AB ductwork, not standby gas filter train, but the direction is to stop venting when containment pressure cannot be maintained below 2 psig. The unprepared applicant may confuse the bases for Securing normal CTMT vent and purge with emergency venting.		
K/A Match The question requires the applicant to have knowledge of the reason for containment venting during high primary containment pressure conditions.		
Technical References: EOP-2 Primary Containment Control step CP-8, 9, EPSTG*0002 pages B-8-20, B-8-14		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-HLO-514 Objective 5: Given an EOP step identify the basis for the action taken		
Question Source:	**Bank # MAR 2014 NRC Q11**	
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(5)&(10)	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295025 (EPE 2) High Reactor Pressure Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: EA1.07 ARI/RPT/ATWS: Plant-Specific	Tier	1
	Group #	1
	K/A	295025 EA1.07
	Rating	4.1
	Revision	1
Revision Statement:		

Question: 50

What is the MINIMUM reactor pressure that would initiate ATWS RPT?

A. 1133 psig

B. 1153 psig

C. 1195 psig

D. 1210 psig

Answer: B

Explanation:

Per R-STM-0053, Reactor Recirculation System, the trip signals to actuate the ATWS interlock are Reactor Vessel Low Level, Level 2 (-43 inches) or Reactor High Pressure (1153 psig).

Distracters:

A. Plausible if applicant confuses with setpoint for actuation of low low set logic from SRV F051C relief setpoint (STM 109 page 61)

C. Plausible if applicant confuses with lowest safety setpoint of 7 of the SRVs. (STM 109 page 61)

D. Plausible if applicant confuses with highest safety setpoint of 4 of the SRVs. (STM 109 page 61).

K/A Match

Technical References: R-STM-0053, Reactor Recirculation System		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-53-LO, Objective (2): Describe the following automatic features and interlocks for the Reactor Recirculation System components, including setpoints, system response and indications (2) ATWS Recirculation Pump Trip (ATWS-RPT)		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b) (5) & (7)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

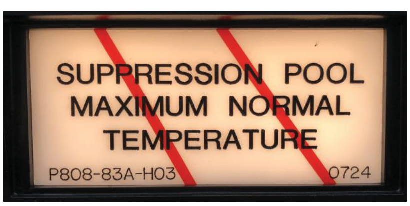
Examination Outline Cross Reference	Level	RO
295026 Suppression Pool High Water Temperature 2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	Tier	1
	Group #	1
	K/A	295026 / 2.4.50
	Rating	4.2
	Revision	3

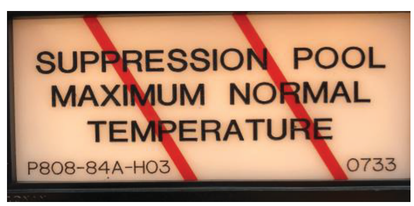
Revision Statement:
Revision 2: Cropped and enlarged recorder images for clarity
Added ", minimum" to each of the answers in part (1)
Added basis for plausibility for 95F to explanation
Relocated (1) and (2) to the front of the question.
Revised stem to indicate that sustained opening of SRV D was performed vice "cycled" because of the resulting supp pool heat up. Also, as well SRV D would not be cycled; SRV's would be opened sequentially.
Added to the explanation why sequential SRV operation is a plausible discriminating distractor.
Revision 3: Added new pictures to make 105F plausible distractor. Also, changed 95F to 105F, to make answers more plausible.

Question: 51

Sustained opening of Main Steam Safety Relief Valve B21-F051D was used for reactor pressure control due to current plant conditions. Below are the indications on H13-P808.

(1) What is the set point for these alarms?
(2) What action should be taken per the Alarm Response Procedure?





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1CMS*TR24A
SUPPR POOL TEMP

1CMS*TR24B
SUPPR POOL TEMP

	(1)	(2)
A.	minimum 105°F	Initiate Suppression Pool Cooling per SOP-31 RHR
B.	minimum 105°F	Perform sequential SRV operation per Op Aid 2000-P-004
C.	minimum 100°F	Initiate Suppression Pool Cooling per SOP-31 RHR
D.	minimum 100°F	Perform sequential SRV operation per Op Aid 2000-P-004

Answer: C

Explanation:

The setpoint for alarms H13-P808-84/84-H03 is ≥100°F. The first time long term action listed in the alarm response procedure is to place RHR into suppression pool cooling per SOP-31 Residual Heat Removal.(ARP- 808-83 page 22, ARP- 808-84 page 26).

Distracters:

A Part 1 is not correct. 105°F is a plausible temperature for this alarm; however, 105°F is the tech spec limit per LCO 3.6.2.1. Suppression pool average temperature shall be ≤105°F when THERMAL POWER is >1% RTP and testing that adds heat to the suppression pool is being performed.. Part 2 is correct.

B Part 1 is not the correct setpoint. Plausibility is given in A. Part 2 Since it was given in the stem that only SRV D had been operated, the suppression pool temperature indication shows localized heating. The distractor is plausible because if the applicant does not recognize that suppression pool cooling is required, he may believe that ARP would have an action to sequentially operate SRVs to equalize indicated temperatures across the suppression pool. (Even though it is the correct action that an operator should always take to evenly heat the suppression pool while using SRVs for pressure control.)

D Part 1 is correct. Part 2 is not a correct action. Plausibility is given in B.

K/A Match

This question requires the applicant to verify the high suppression pool temperature alarm setpoint and determine which controls to operate per the ARP to mitigate the high temperature..

Technical References:

ARP- 808-83/84-H03, Op Aid 2000-P-004

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-57-LO, Objective 8: Describe the primary containment interface(s) with the following systems: (8)
Residual Heat Removal (RHR)

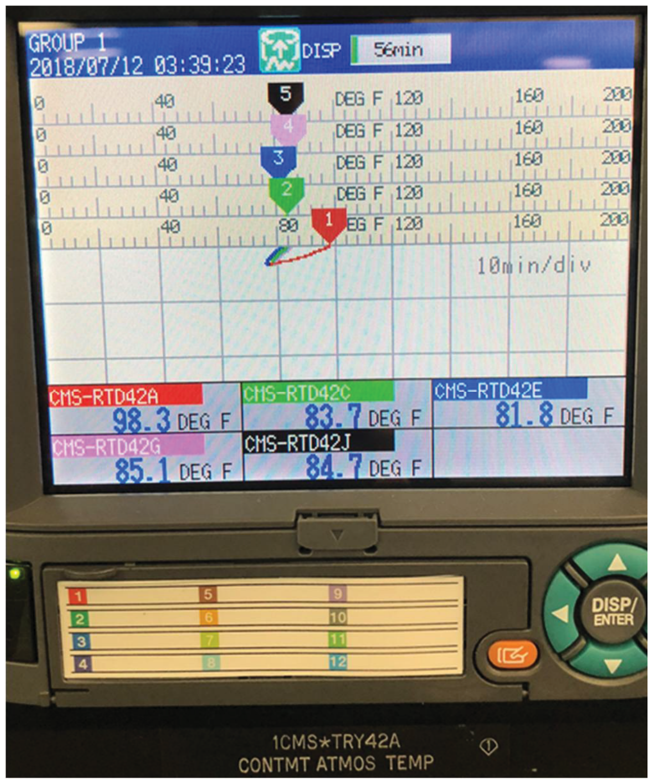
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(10)	
Level of Difficulty:	3	

SRO Only Justification: NA

PRA Applicability:

Examination Outline Cross Reference	Level	RO
295027 High Containment Temperature (Mark III Containment Only) EA2. Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) : EA2.03 Reactor pressure: Mark-III	**Tier**	1
	Group #	1
	K/A	295027 EA2.03
	Rating	3.3
	Revision	2
Revision Statement: Deleted Point 1 from picture and deleted reference to Point 1 in the stem.		

Question: 52



An ATWS is in progress.

Based on the indications, EOP-2, Primary Containment Control, ___(1)___ required to be entered.

Based on Containment Temperature trend, per EOP-1A, RPV Control - ATWS, Emergency Depressurization ___(2)___ be anticipated.

	(1)	(2)
A.	is	can
B.	is	cannot
C.	is not	can
D.	is not	cannot

Answer: B

Explanation:

Per EOP-2, entry condition for Containment Temperature is above 90°F.

Per EOP-1A, RPV Control – ATWS, Emergency Depressurization cannot be anticipated. Applicant may confuse EOP-1 and EOP-1A rules for anticipating emergency depressurization.

Distracters:

(1) Per EOP-2, Suppression pool temperature above 100F is entry condition. It is plausible if applicant confuses suppression pool temperature entry condition with Containment temperature entry condition.

(2) Per EPSTG*0002, bases for EOP-2, Step (CT-4, CT-5) states override two in EOP-1 Step RP-1 permits rapid depressurization through the main turbine bypass valves and main steam line drains in anticipation of emergency RPV depressurization. Plausible to confuse EOP-1 and EOP-1A pressure control strategies.

It is plausible for the applicant to not enter EOP-2 until after 100F and anticipate emergency depressurization based on the continued upper temperature trend.

K/A Match

Applicant must have knowledge of EOP-2 entry requirement for Containment Temperature. The applicant must also be able to determine reactor pressure control per EOP-1A pressure control leg, as it applies to high containment temperature trend.

Technical References:

EOP-1, RPV Control

EOP-1A, RPV Control – ATWS EOP-2, Primary Containment Control EPSTG*0002, bases for EOP-2		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-LO-514, Objective 4: Analyze Control Room indications to determine if EOP-2 entry is required		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(10)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	RO
295030 Low Suppression Pool Water Level Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: EK2.08 SRV discharge submergence	**Tier**	1
	Group #	1
	K/A	295030 / EK2.08
	Rating	3.5
	Revision	3
Revision Statement: Revision 2: Changed A to 12’3” and moved other values down and deleted 21’6”With the revised distractors,12’3” is the lowest suppression pool level of the selections; however, it is a plausible distractor since it would cause direct pressurization of the containment air space, but it is an incorrect answer because the question is asking for the suppression pool level needed for SRV operation. Revision 3: Added “maximum” and ”commence” to the stem to eliminate subset of A.		

Question: 53

For SRV operations, at which of the following maximum suppression pool levels will direct pressurization of containment air space commence?

A. 12 feet, 3 inches

B. 13 feet

C. 14 feet

D. 15 feet, 5 inches

Answer: B
Explanation: This is the elevation of the top of the SRV discharge device below which opening of an SRV may cause pressurization of the containment air space
Distracters: A. 12’3” is the level of first (top) drywell horizontal vent which would cause a loss of pressure suppression function.**Z**12’3” is the lowest suppression pool level of the selections; however, it is a plausible distractor since it would cause direct pressurization of the containment air space, but it is an incorrect answer because the question is asking for the suppression pool level needed for SRV operation. C 14’ is the Vortex limit discussed in Caution #5 of the EOPs. (EPSTG-0002 page A-58 D 15’5” is 2 feet above the horizontal vents; this level is associated with a leak from the DW passing through the horizontal vents. (EPSTG-0002 page B-8-32)

K/A Match This question requires the applicant to have knowledge of which low suppression pool is required for proper SRV discharge submergence.		
Technical References: EOP-1, 2, EPSTG-0002 pages B-6-64, B-8-32, B-8-31 ANDA-58 RBS STM-0057 Figure 2 Weir Wall Cutaway		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RPPT-OPS-HLO-511 Objective 5: Given plant parameters or indications, determine EOP/SAP Cautions applicability.		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295031 Reactor Low Water Level EK3. Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL : EK3.01 Automatic depressurization system actuation	Tier	1
	Group #	1
	K/A	295031 EK3.01
	Rating	3.9
	Revision	2
Revision Statement: Added, “Per USAR Section 6.3, the reason … “ to the stem.		

Question: 54

Per USAR Section 6.3, 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 experienced by the reactor coolant pressure boundary

Answer: A

Explanation:

Per R-STM-202, Automatic Depressurization System, the purpose of the Automatic Depressurization System is to provide automatic depressurization of the reactor pressure vessel by activating and opening seven of the sixteen Safety Relief Valves (SRVs). This depressurization is intended to lower RPV pressure to the point that Residual Heat Removal - Low Pressure Coolant Injection mode (RHR - LPCI) and Low Pressure Core Spray (LPCS) systems can inject into the RPV to flood the core.

Distracters:

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.

K/A Match

Applicant must have required knowledge for the reason for ADS actuation which is to recover low reactor water level.

Technical References:

RLP-STM-0202, ADS

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0202 Objective 1: State the purpose of ADS (1)

Question Source:	**Bank # March 2014 NRC Q16**	X
	Modified Bank #	
	New	

Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	

10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	2	

SRO Only Justification:

N/A

PRA Applicability:

Top 10 Risk Significant Systems: SRV Depressurization

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="6">295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN :
EA1.10 Alternate boron injection methods: Plant-Specific.</td><td>Tier</td><td>1</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>295037 / EA1.10</td></tr>
<tr><td>Rating</td><td>3.7</td></tr>
<tr><td>Revision</td><td>3</td></tr>
<tr><td colspan="3">Revision Statement:
Deleted raising or lowering reactor pressure from the distractors. Replaced with distractors for whether or not SLC is there is not injecting.
Added trends with the given parameters and high drywell pressure to meet Table L-5 conditions.
Added details to the explanation to support changes in the stem.
Rev 3: Changed C and D part (2) to maintained to make more plausible.</td></tr>
</table>

Question: 55

The CRS has directed that Enclosure 15, ALTERNATE SLC INJECTION be implemented per EOP-1A, RPV Control - ATWS.

Current plant conditions are:

• Reactor power	25% steady
• Reactor level	-95” steady
• Reactor pressure	942 psig steady
• SLC hydro pump discharge	846 psig steady
• Suppression pool level	18’-10” steady
• Suppression pool temperature	115°F rising slowly
• Drywell pressure	2 psig rising slowly

With current conditions:

(1) alternate SLC __________ injecting into the reactor.
(2) Reactor level should be __________.

	(1)	(2)
A.	is	lowered
B.	is not	lowered
C.	is	maintained

D. is not maintained

Answer: B
Explanation: The question supports the KA for the ability to monitor Alternate Boron Injection during an ATWS by determining whether or not conditions (reactor pressure versus SLC pump pressure) are present to support injecting SLC into the reactor. Step 3.8.3 of Enclosure 15 requires that alternate SLC hydro pump discharge pressure be above reactor pressure to support SLC injection into the reactor. Since given reactor pressure is greater than SLC pump discharge pressure, SLC “is not” injecting into the reactor. The conditions given in the stem meet all of the conditions of Table L-5 of EOP-1A. With all of the conditions met in Table L-5, EOP-1A step RLA-14 requires lowering reactor level until reactor level reaches -100 inches OR DW pressure is less than 1.68 psig OR reactor power is less than 5%. Since and drywell pressure and reactor power will not be corrected prior to reaching -100”, level will need to “lowered” to -100”
Distracters: A Part 1 is incorrect because Hydro pump discharge pressure is less than reactor pressure, but is plausible if an applicant misinterprets the given Hydro pump discharge pressure and reactor pressure . Part 2 is correct as given in B C Part 1 is incorrect as given in A. Part 2 is incorrect because reactor level needs to be lowered not maintained. RPV water level would be maintained -60 to -140 inches if level/power condition were not met. Per EOP-1A, step RLA-15, continue lowering RPV level until any of the following: Reactor power drops below 5% (power is currently 25%), RPV level reaches -100 inches (currently -95 inches), or All SRVs remain closed and DW pressure remains below 1.68 psid (currently 2 psid and rising slowly). Maintaining level is plausible if applicant confuses level/power requirements. D Part 1 is correct as given in B. Part 2 is incorrect as given in C.
K/A Match This question determines the applicants ability to monitor parameters during alternate SLC injection in ATWS conditions
Technical References:

EOP-1A steps RLA-9, RLA-14 and RPA-3, EOP-5 Enclosure 15 section 3.8 on page 56		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-HLO-516-ILO Objective 1: Given the applicable Enclosure, and Flowcharts, determine the purpose, method of implementation, and resulting system response for each enclosure (1).		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">295038 High Off-Site Release Rate
EA2. Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE :
EA2.03 †Radiation levels</td><td>Tier</td><td>1</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A</td><td>295038 EA2.03</td></tr>
<tr><td>Rating</td><td>3.5</td></tr>
<tr><td>Revision</td><td>1</td></tr>
<tr><td colspan="3">Revision Statement:</td></tr>
</table>

Question: 56

Plant is operating at 100% power.

A Loss of Turbine Building Ventilation occurs.

A significant leak occurred on the steam supply to MSR#1.

The CRS has directed the ATC operator to place the mode switch in SHUTDOWN. Control Rods failed to insert. EOP-1A execution is in progress. The following conditions exist:

- Reactor power: 17%
- MSIVs open

- RMS-RE125 MAIN PLANT EXHAUST DRMS Green status
- RMS-RE110 AUX BLDF VENTILATION DRMS Green status
- RMS-RE118 TURBINE BLDG VENT DRMS Blinking Magenta status

Emergency Response Organization has been activated.

Offsite release teams have reported 850 mR/hour at the site boundary.

Which of the following accurately describes the current condition?

A. An unfiltered, monitored release is in progress.

B. An unfiltered, unmonitored release is in progress.

C. A filtered, monitored release is in progress.

D. A filtered, unmonitored release is in progress.

Answer: B		
Explanation: A leak in the MSR area producing 850mR/hr at the site boundary should be observed on RMS-RE118 and RMS-RE125. Since these monitors are not in alarm, the release is unmonitored. A leak outside secondary containment is unfiltered.		
Distracters: A. Plausible if applicant confuses with fuel building HVAC. The normal exhaust is unfiltered and monitored. C. Plausible if applicant confuses with standby gas treatment release path. The exhaust if filtered and monitored in the main plant exhaust. D. Plausible if applicant recognizes the release is not monitored, but does not understand the flow path from the MSR to the site boundary.		
K/A Match Applicant must have knowledge to determine a flow path from MSR to site boundary. Applicant must also interpret the given indications, to understand the release is not monitored.		
Technical References: PID22-03A, PID-22-01C		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-409-LO Objective 2: Describe the basic operation/draw a simplified flowpath of the Auxiliary Building HVAC System during the following (2) LOCA or Auxiliary Building Exhaust High Radiation Condition		
Question Source:	**Bank # 2008 NRC Q18**	X
	Modified Bank #	

	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(11) & (13)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
600000 Plant Fire On Site Knowledge of the interrelations between PLANT FIRE ON SITE and the following: AK2.01 Sensors / detectors and valves	Tier	1
	Group #	1
	K/A	600000/ AK2.01
	Rating	2.6
	Revision	2
Revision Statement: Revised stem to “A pre-action sprinkler system is comprised of _______ “ Added “a” network… (A&B) and “a” set…(D)		

Question: 57

A pre-action sprinkler system is comprised of ______________________.

A. a network of dry piping containing open spray heads

B. a network of pressurized water filled piping containing heat sensitive closed sprinkler heads

C. a set of thermally sensitive fusible sprinkler heads with a deluge valve which opens based on fire or smoke detectors

D. a set of fixed-temperature detectors that actuate the Discharge control heads which pressurizes the manifold causing the main discharge seat to open

Answer: C
Explanation: The stem of the question asks how does a pre-action sprinkler system will operate during a fire on site and this system uses a unique combination of detectors and valves to operate(STM-250 page 16). The other types of fire protection systems, given as distractors, operate differently than this system and the unprepared applicant may confuse these systems.
Distracters: A This is the description of a deluge system(STM-250 page 17) B This is the description of a Wet pipe system(STM-250 page 14) D This is the description of a Carbon Dioxide System(STM-250 page 23)
K/A Match

This question tests the applicants' knowledge concerning fire detectors and valves as they interrelate to a fire on site.		
Technical References: STM-250 Fire Protection and Detection		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-250-LO Objective 3: Describe the basic operation of the Fire Detection Supervisory System including the types and operation of the detectors. (3)		
Question Source:	**Bank #**	2008 audit
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
t10CFR Part 55 Content:	41(b)(4)	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances AK1. Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AK1.03 Under-excitation	Tier	1
	Group #	1
	K/A	700000 AK1.03
	Rating	3.3
	Revision	2
Revision Statement: Added "while paralleled to the grid" after "Generator" to clarify that C is incorrect. Revised C to "in-plant motors". Lowercased A,B,&D.		

Question: 58

Under-excitation of the Main Generator while paralleled to the grid results in ______ overheating.

A. turbine blading

B. generator field

C. in-plant motors

D. generator armature

Answer: D
Explanation: Per R-STM-310, Main Generator and Alterrex Exciter, operating the generator in an under-excited state can lead to armature overheating and rotor instability.
Distracters: A. Per R-STM-310, fThe generator is protected against anti-monitoring by circuit 1SPUN04. Motoring of the generator occurs when the steam flow to the turbine is reduced to less than no-load flow when the generator is still on the line. Under these conditions, the generator operates as a synchronous motor driving the turbine. The turbine can be subjected to overheating as a result of motoring. B. Per R-STM-310, the maximum excitation limiter is designed to protect the generator field from overheating due to prolonged over-excitation. C. Although operating in an under excited condition reduces the generator output

voltage which by itself would cause lower bus voltage and higher motor currents and overheating; however, in plant bus voltage is determined by grid voltage which will prevent any significant drop in bus voltage by the grid supplying reactive current to the main generator.

K/A Match

Applicant must understand the operational implications to the main generator while operating in under-excitation condition.

Technical References:

R-STM-310, Main Generator and Alterrex Exciter

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0310, Objective 3: Describe the Exciter – Regulator Electrical Flow Path (3)

Question Source:	**Bank # Dec 2014 NRC Q20**	X
	Modified Bank #	
	New	

Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	

10CFR Part 55 Content:	41(b)(5)	
Level of Difficulty:	3	

SRO Only Justification:
N/A

PRA Applicability:

Examination Outline Cross Reference	**Level**	RO
295008 High Reactor Water Level Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL : AK1.03 Feed flow/steam flow mismatch.	**Tier**	1
	Group #	2
	K/A	295008 / AK1.03
	Rating	3.2
	Revision	2
Revision Statement: Changed the stem to indicate that the high water level alarm (which is received at 38 inches) is sealed in. Therefore, distractors A, B, and C would be incorrect because they would not result in a steady-state reactor level 5 inches above the automatic level setpoint. Only the feed flow instrument failure would result in a 5 inch steady-state level rise.		

Question: 59

Current plant conditions are as follows:

- Reactor Power is 50%.
- Reactor water level is 41" and stable
- Feedwater Level Control in 3-ELEMENT
- Master Level Controller set at 36" in AUTO

The the following alarm is currently sealed in:

Which of the following failures would cause this indication?

A. One SRV has failed open.

B. One Startup Feedwater Regulating Valve has failed open.

C. One Turbine bypass valve has failed open.

D. One feed flow instrument has failed downscale.

Answer: D

Explanation:
To support the KA, the only way to have an operational implication from a feed flow steam flow mismatch is from a sustained feed flow steam flow mismatch caused by a failure. When a feed flow instruments fails low, and steam flow remains the same, the Feedwater Control System, while in 3-element, will open the FRV to null the steam flow/ feed flow mismatch. As the FRV opens reactor level will rise. Once the reactor level exceeds the master level control set point of 36 " the level dominant control system will halt the FRV opening and level will stabilize at a new higher value.(STM-107 page 57)

Distracters:
A. An open SRV will initially cause steam flow to increase; however, the turbine control valves will respond to the pressure decrease and return the total steam flow to the original value. The SRVs divert steam prior to the main steam flow sensors resulting in sensed steam flow vs the feedwater system being lower. Therefore, the FRV's initially close, reactor level lowers, and then stabilizes at a slightly lower steady-state level. The unprepared applicant may not understand the effect of an open SRV on the feedwater control system at power.
B. A failed open startup FRV would cause level to initially start to rise; however, the level would see the increase in feed flow and close the main FRV to maintain feed flow and reactor stable. The unprepared applicant may not understand that the feed flow from the startup FRV is included in the 3 element control circuit.(STM-107 page 57)
C. A Turbine bypass valve failing open would cause an immediate response from the turbine control valves closing to cause negligible change in total steam flow and a resulting negligible change in reactor water level.. The unprepared applicant may not understand the plants response to a bypass valve failing open at 60% power.

K/A Match
This question tests the applicants' knowledge of how a feed flow / steam flow mismatch will impact reactor level control system to cause a high water level condition.

Technical References:
RLP-STM-107-FWLC-LO

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:
RLP-STM-107-FWLC-LO Objective 13: Given a set of plant conditions, predict the effect that a loss or malfunction of the Feedwater Level Control System will have on the following: (13)

Reactor water level		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	CFR 55.41(b)4	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

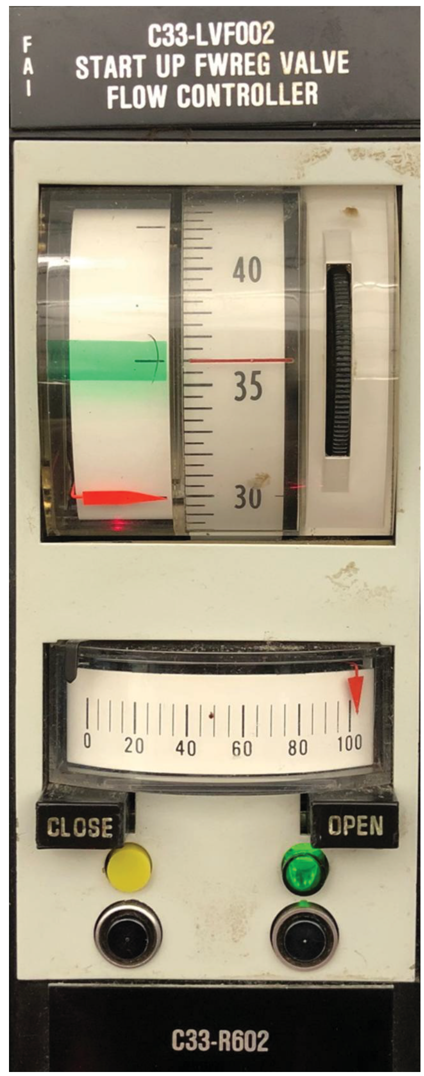
Examination Outline Cross Reference	Level	RO
295009 Low Reactor Water Level AK2. Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: AK2.02 Reactor water level control	Tier	1
	Group #	2
	K/A	295009 AK2.02
	Rating	3.9
	Revision	2
Revision Statement: Removed extra “Due to”.		

Question: 60

Based on the following indications:

The alarm is due to ___(1)___ RPV water level.

The C33-LVF002, START UP FWREG VALVE FLOW CONTROLLER, ___(2)___ responding properly.



	(1)	(2)
A.	high	is
B.	high	is not
C.	low	is
D.	low	is not

Answer: C
Explanation:

The alarming condition must be clarified using the controller indication. The top left indication is the deviation meter between the setpoint and actual level. The setpoint is at 36 inches. The deviation is below the setpoint, indicating a low level condition.

The horizontal indication is used for output demand. The demand is at 100, which gives the valve an open signal. In a low level condition the valve should open. This is the correct response of the feedwater level control system.

Distracters:

The high level alarm indication is plausible if the applicant confuses the vertical indication for the demand. If the valve was given a low demand or demand to close the valve that would indicate a high water level condition.

If the applicant assumes the water level is high the demand should be zero to close the valve.

If the water level is low and the applicant confuses the vertical indication as demand signal, the applicant would assume the valve is getting a closed signal.

K/A Match

Applicant must interpret the feedwater level controller to determine actual low reactor water level condition and interpret correct controller response to the low water level condition.

Technical References:

Per R-STM-0107-FWLC-LO, Feedwater level Control Systems

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-107, FWLC, Objective (5): Describe the purpose and operation of the following Feedwater Level Control System components: (5)

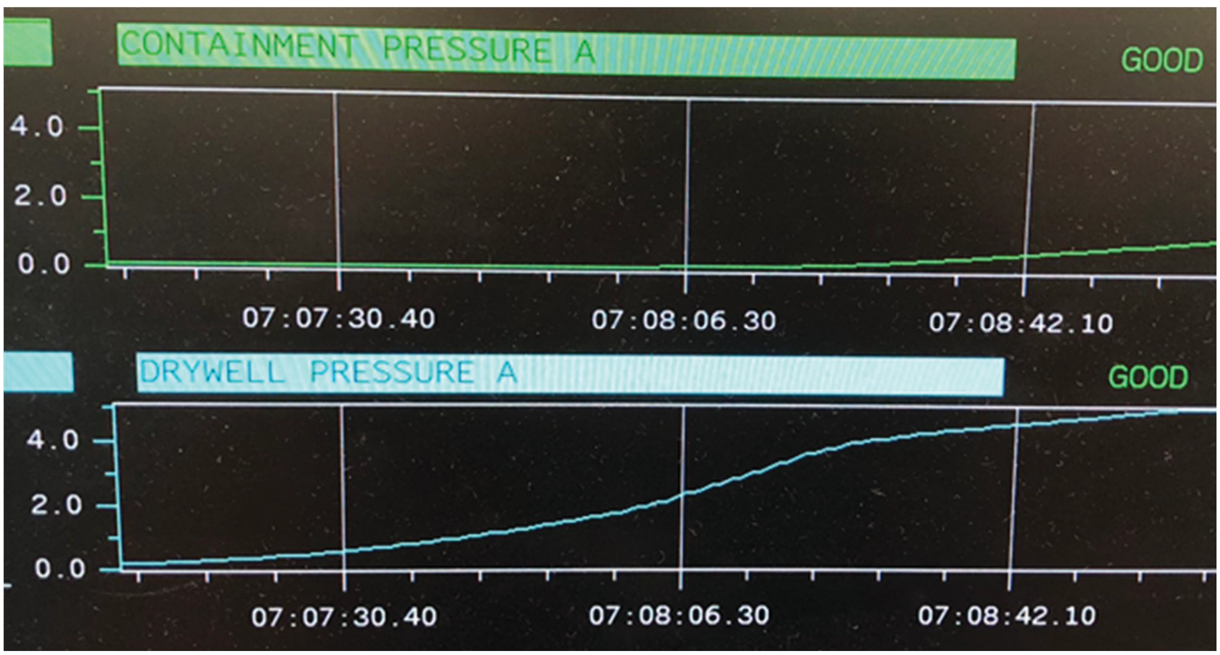
2018 RBS NRC Examination

Startup Reg Valve Controller		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295010 High Drywell Pressure Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : AK3.01 Drywell venting	**Tier**	1
	Group #	2
	K/A	295010 / AK3.01
	Rating	3.8
	Revision	3
Revision Statement: Revision 2: NRC assigned a new KA for this question. Replaced question to support the new KA. Revision 3: Rephrased stem to eliminate two part question.		

Question: 61

What are the reasons for the indications below?



A. A steam leak in the drywell with a concurrent containment steam leak.

B. Steam leak in the drywell ONLY and the Drywell horizontal vents uncovered.

C. Drywell horizontal vents uncovered with a Loss of Containment cooling ONLY

D. A Loss of Drywell cooling with a concurrent loss of Containment cooling

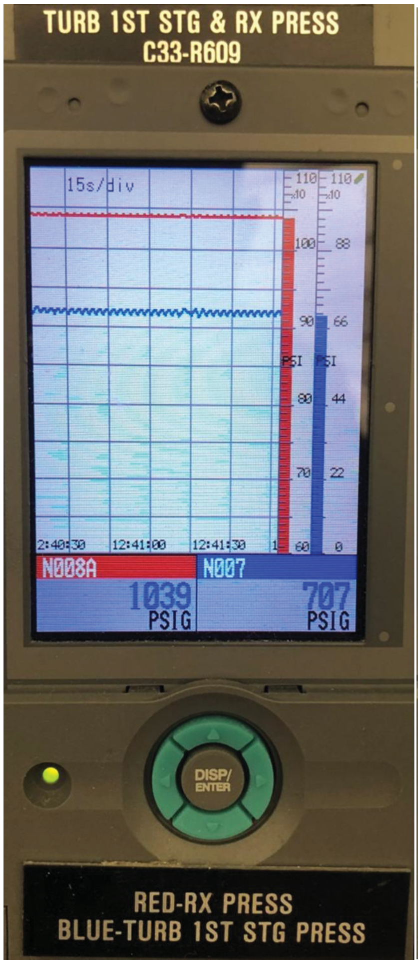
Answer: B
Explanation: As indicated in the recorder trace, drywell pressure is slowly rising due to a “steam leak in the drywell”. When drywell pressure is greater than 2 psig, the first row of “drywell horizontal vents become uncovered” releasing steam and non-condensibles through the suppression pool. The steam is condensed and the non-condensibles are released to the containment atmosphere causing a rise in containment pressure. Mark III drywell venting explanation from the System Training Manual: A buildup of steam pressure in the Drywell forces the water in the wetwell, area between the Drywell wall and the Weir Wall, down until the water is depressed to the top of the first row of horizontal vents (top is 13’5”) and allows the steam to be vented into the Suppression Pool and condensed. If the pressure in the Drywell is high enough, the water level in the wetwell is depressed to the second and third rows of vents. If initial Suppression Pool level is 19’6”, a Drywell to Containment D/P of approx. 2.6 psid starts to uncover the first row of horizontal vents. (STM-0057 page 10)
Distracters: A It is plausible that a steam leak in the drywell and a concurrent steam leak in the containment could cause both DW and Containment pressures to rise. However, the recorder trace shows a Containment pressure response characteristic of non condensibles from the suppression pool as indicated by a constant DP between the DW and Containment. C It is plausible that a loss of Containment cooling would cause a rise in containment pressure followed by a rise in drywell pressure (Containment venting into the drywell) However, if this occurred the Containment pressure would lead DW pressure and would occur over a longer time frame. D It is plausible to lose drywell cooling with a concurrent loss of containment cooling resulting in a recorder trace similar to that which is given. However, based on Loss of Drywell Cooling Calculation (page 5) it would take approximately 21 minutes for the drywell to reach 2 psig. The provided recorder trace occurs in a span of only approximately 2 minutes.
K/A Match: Applicant must understand the reason that drywell venting occurs in a Mark III containment with a small steam leak in the drywell. The KA is supported by the applicant interpreting drywell and containment pressure response which represents drywell venting in a Mark III containment.

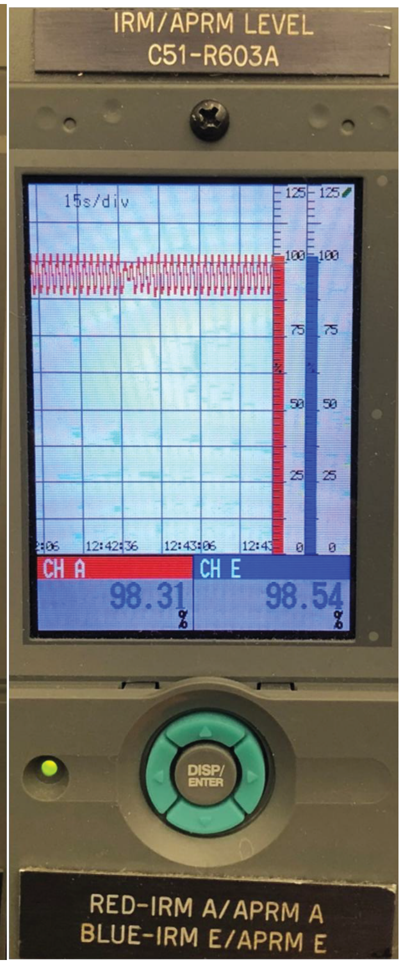
Technical References: R-STM-0057 Primary Containment and Aux Drywell Temperature and Pressure Transients Following a Loss Offsite Power and/or Loss of Drywell Cooling Events calculation 1986		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-57-LO Objective 3: Describe the purpose, construction, and operation of the following components and/or subsystems: (3) Drywell		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	CFR 55.41(b)10	
Level of Difficulty:	2	
SRO Only Justification: NA		
PRA Applicability:		

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="6">295014 Inadvertent Reactivity Addition
AA1. Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION:
AA1.06 Reactor/turbine pressure regulating system</td><td>Tier</td><td>1</td></tr>
<tr><td>Group #</td><td>2</td></tr>
<tr><td>K/A</td><td>295014 AA1.06</td></tr>
<tr><td>Rating</td><td>3.3</td></tr>
<tr><td>Revision</td><td>3</td></tr>
<tr></tr>
<tr><td colspan="3">Revision Statement:
Revision 2: Revised stem to say “What is the first operator action….” to distinguish it as the only correct answer. Replaced distractor C with a distractor based on a supplemental action from AOP-24 Thermal Hydraulic Stability Controls which is discriminating for an applicant believes the indications are the onset of thermal hydraulic instability. Deleted the “if- then” format from distractor D. Paraphrased the answer so does not sound like a cut-and-paste out of a procedure and was consistent with the format of the other distractors.
Revision 3: Revised D to include wording from GOP-5 step for adjusting pressure set.</td></tr>
</table>

Question: 62

While performing OSP-102, TURBINE VALVE TESTING, the following indications are observed:





What is the first operator action that should be taken based on these indications?

A. Place the Reactor Mode Switch to SHUTDOWN

B. Stop all plant operations that have a potential to change turbine load or RPV pressure.

C. Insert control rods using the Shutdown Control Rod Sequence Package.

D. Adjust Press Set using the Reactor Pressure Set Calculation and Adjustment graph.

Answer: A
Explanation: Per the indications provided, reactor pressure and turbine 1st stage pressure is oscillating about 5 psig due to malfunction. APRM oscillations are about 15% peak to peak swings. The oscillating reactor power represents a series of inadvertent reactivity additions due to a turbine pressure control malfunction in support of the KA.

The applicant needs to interpret the control room indications to determine that the reactor power oscillations are in excess of the action level requiring a reactor scram.

Per AOP-17, REACTOR PRESSURE CONTROL MALFUNCTIONS, Immediate Operator Actions, IF APRM readings are observed to be oscillating with sustained peak-to-peak swings of greater than 10% rated power, THEN place C71A-S1, REACTOR SYSTEM MODE SWITCH, to SHUTDOWN.

Distracters:

B. Per AOP-17, supplemental actions, 5.5.5. IF all three main steam throttle pressure transmitters FAIL AS IS, THEN perform the following: 1. Stop all plant operations that have a potential to change turbine load or RPV pressure.

This supplemental action is plausible because with oscillations, if an applicant does not recognize the requirement to scram the reactor, stopping plant operations that can change turbine load or RPV pressure could be an effective mitigating action.

C. Per supplemental actions. 5.1.1. IF entry into the Restricted Region has occurred AND the reactor mode switch has not been placed in the SHUTDOWN position, THEN immediately exit this region by performing the following actions:

- Insert Control Rods using the Shutdown Control Rod Sequence Package or as specified by Reactor Engineering, OR
- Raise recirculation flow by opening Recirc FCVs

Inserting control rods is plausible if the applicant mistakenly interprets the indications as thermal hydraulic instability. However, the action for inserting rods would only be applicable if entering the restricted with the reactor 100% power, reactor power/core flow are not in the restricted region.

D. Per GOP-5, Power Maneuvering, Attachment 1, 2 IF returning the pressure set to the Nominal Press, THEN perform the following:

2.1 Determine the nominal pressure for the current reactor power (CTP % Rated), using the Reactor Pressure Set Adjustment graph on the next page of this Attachment.

K/A Match

Applicant must interpret the indications provided and determine the correct action as it applies to the reactivity addition due to pressure changes.

Technical References:

AOP-17, REACTOR PRESSURE CONTROL MALFUNCTIONS

AOP-24, THERMAL HYDRAULIC STABILITY CONTROLS GOP-5, Power Maneuvering		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-509-ILO EHC, Objective 9: Describe the Turbine EHC response and parameters to monitor from the Main Control Room during the following: Turbine roll Reactor pressure changes Turbine and Bypass valve testing		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(5)&(10)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295033 High Secondary Containment Area Radiation Levels Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : EA2.03 †Cause of high area radiation	Tier	1
	Group #	1
	K/A	295033 / EA2.03
	Rating	3.7
	Revision	1
Revision Statement:		

Question: 63

The plant is operating at 100% power when an abnormal condition occurs.

Based on the control panel indications below:

What abnormal event has occurred?

A. Steam leak in the RCIC room

B. Suction line leak in the RCIC room

C. RCIC oil cooler water leak

D. RCIC pump seal leak

Answer: A
Explanation: The applicant must first recognize that the control panel indications are for plant radiation and radioactive release levels (as opposed to sump levels or temperature values). Once the applicant recognizes the control panel indications as radiation levels, he needs to reason what is causing the high radiation levels. The high RCIC area radiation level is caused by a steam supply leak. Steam supply to the RCIC turbine is from main steam which is highly radioactive at 100% power due to hydrogen injection. This is also evident by the rise in main plant exhaust stack gaseous monitor readings(RE125). EOP-3 Secondary Containment and Radiation Release Control uses table SC-2 to define the area of concern for entry conditions.(EOP-3 page B-9-3)
Distracters: B A applicant may believe this to be the leak source because the suppression pool is a suction source for RCIC and the water is contaminated. However that water would not cause area radiation monitor alarm and would not propagate as a gas to the main plant exhaust.(STM-209 page 5) Also, the applicant may assume that the abnormal indications for RCIC is a sump level and not a radiation level. C A applicant may believe this to be the leak source because the suppression pool is a suction source for RCIC and the water is contaminated and the RCIC pump discharge is used as the cooling water for the oil cooler. However that water would not cause area radiation monitor alarm and would not propagate as a gas to the main plant exhaust.(STM-209 page 15 and 54) Also, the applicant may assume that the abnormal indications for RCIC is a sump level and not a radiation level. D A applicant may confuse the pump seal with the turbine seal. The turbine seal would be a source of radioactive steam; however, the pump seal would leak water from the suction source, CST or suppression pool, which would not cause area radiation monitor alarm and would not propagate as a gas to the main plant exhaust. Also, the applicant may assume that the abnormal indication for RCIC is a sump level and not a radiation level.
K/A Match This question requires the applicant to determine the cause of the high area radiation in secondary containment RCIC room.

Technical References: STM-209 pages 5, 15, 54 EPSTG-2 page B-9-3		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-209-LO Objective 11: Given a set of conditions and drawing of the controls, instrumentation, and/or alarms located in the Main Control Room, identify the status of the RCIC System by evaluation of the controls/instrumentation/alarms. (11)		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	CFR 55.41(b)11	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
295034 Secondary Containment Ventilation High Radiation 2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	Tier	1
	Group #	2
	K/A	295034 2.4.2
	Rating	4.5
	Revision	2

Revision Statement:

Revised Part 1 to delete reference to EOP-1. Added AOP as a distractor. AOP added is AOP-3 AUTOMATIC ISOLATIONS. This is a credible distractor since the automatic actions of SGT initiation results in an entry condition to this AOP. Also, revised Part 2 distractors B&D to “Standby Gas Treatment System auto initiates” (previously stated SGT fan A starts). The high radiation condition does cause SGT fan A to start; however, additionally dampers realign and due to a low flow condition (described in the alarm response procedure) SGT fan B also starts. The phrase “Standby Gas Treatment System auto initiates” encompasses the damper realignment and both SGT fans starting. “Standby Gas Treatment System auto initiates” is a cut-and-paste of the auto response given in AOP-3.

Revised the explanation to incorporate AOP-3 as a credible discriminatory distractor. With the question revision, the applicant now needs to determine whether the given condition is an EOP-3 entry condition and the correct automatic response to that condition.

Added AOP-3 excerpt as a reference to support the revised question.

Question: 64

Upon the receipt of the following alarm and validated above high setpoint:

(1) What procedure(s) must be entered based on receipt of valid alarm ONLY?
AOP-3 AUTOMATIC ISOLATIONS
EOP-3, SECONDARY CONTAINMENT and RADIOACTIVITY RELEASE CONTROL

(2) What automatic action(s) should have occurred?

	(1)	(2)
A.	AOP-3 ONLY	HVN, HVR, SWP valves isolate
B.	AOP-3 ONLY	Standby Gas Treatment System auto initiates
C.	AOP-3 and EOP-3	HVN, HVR, SWP valves isolate
D.	AOP-3 and EOP-3	Standby Gas Treatment System auto initiates

Answer: **D**

Explanation:

Per AOP-3, Alarm P863-71A-C07 above the high setpoint causes dampers to realign as part of Standby Gas Treatment System initiation which is a “Symptom” for AOP-3 requiring entry into AOP-03. Per EOP-3, Alarm P863-71A-C07 is indication of radiation levels above the max normal level, which is entry into EOP-3.

Per OSP-22, 5.1.2 The red and/or blue diagonal lines on selected annunciator windows, in the Control Room, are installed as a visual aid. The red diagonal lines denote annunciators requiring review of the EOPs for possible entry conditions. Applicant must interpret the alarm indication as an entry into EOP-3.

Per H13-P863/71A/C07, DIV I ANNULUS EXH RADN ALARM, at the High Alarm setpoint the dampers realign and both Standby Gas Treatment fans start. (i.e. Standby Gas Treatment System auto initiates)

Distracters:

(1) For A & B which are incorrect, AOP-3 is plausible if applicant does not recognize alarm condition as an entry condition into EOP-3.

(2) For A & C are incorrect, but plausible if the applicant confuses Containment to Annulus Differential Pressure High Alarm with Div 1 Annulus Exh Radn alarm . Per EOP-3, Automatic Isolations and AOP-03, Containment to Annulus Differential pressure – High: -12 inches H2O causes HVN, HVR, and SWP valves to isolate.

K/A Match

Based on the Division 1 Annulus Exhaust Radiation Alarm, the applicant must have the knowledge to determine the EOP entry setpoint and automatic action associated with it.

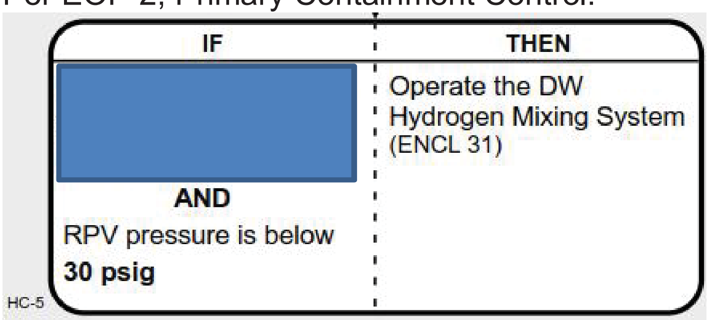
Technical References: AOP-3 AUTOMATIC ISOLATIONS EOP-3, SECONDARY CONTAINMENT and RADIOACTIVITY RELEASE CONTROL H13-P863/71A/C07, DIV I ANNULUS EXH RADN ALARM		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-LO-515, Objective 3: Determine if entry is required based on Control Room indication RLP-STM-511-LO, Objective (7): Describe the interrelationship(s) between the Radiation Monitoring System and the following systems: (7) Reactor Building Ventilation		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	41(b)(11)	
Level of Difficulty:	3	
SRO Only Justification: N/A		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
500000 High Containment Hydrogen Concentration Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS: EK1.01 Containment integrity	Tier	1
	Group #	2
	K/A	500000 / EK1.01
	Rating	3.3
	Revision	2
Revision Statement: Revised C and D to "the Drywell Edited stem to: "Per the EOP-2 bases, prior to operating the drywell hydrogen mixing system, RPV pressure must be below the ___(2)___ to ensure that any RPV blowdown bypassing …"		

Question: 65

The drywell hydrogen mixing system equipment is located in ___(1)___.

Per EOP-2, Primary Containment Control:



Per the EOP-2 bases, prior to operating the drywell hydrogen mixing system, RPV pressure must be below the ___(2)___ to ensure that any RPV blowdown bypassing the suppression pool will not threaten containment integrity.

HDOL - Hydrogen Deflagration Overpressure Limit
PCPL - Primary Containment Pressure Limit

	(1)	(2)
A.	Containment	HDOL
B.	Containment	PCPL
C.	The Drywell	HDOL

D. The Drywell PCPL

Answer:	**B**

Explanation:

Since drywell hydrogen mixing system equipment is located in the containment and is capable of providing a spurious ignition source, operation of the drywell hydrogen mixing system must be dependent upon the concentration of containment hydrogen with respect to the containment Hydrogen Deflagration Overpressure Limit (HDOL). The containment HDOL is the highest containment hydrogen concentration at which a deflagration will not generate pressures in excess of the structural capability of the containment.

The drywell hydrogen mixing system allows direct communication between containment and drywell atmospheres without forcing non-condensable gas through the suppression pool, as such RPV pressure must be below the Primary Containment Pressure Limit (PCPL- 30 psig) prior to operating the drywell mixing systems. This restriction ensures RPV pressure is sufficiently low that any RPV blowdown bypassing the suppression pool will not threaten containment integrity.

Distracters:

(1) Drywell is plausible if applicant does not know where the drywell hydrogen mixing equipment is located.

(2) HDOL is plausible if applicant misunderstands the requirement to operate the DW Hydrogen Mixing System to be the Hydrogen Deflagration Overpressure Limit. HDOL is a hydrogen based limit, but it is not the applicable curve per EOP-2 bases for this step to operate Enclosure 31.

K/A Match

The applicant must have knowledge of EOP bases and the operational implication of violating the PCPL curve, which threatens containment integrity.

Technical References:

EOP-2, Primary Containment Control

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective: RLP-LO-514 Objective 5: Given an EOP step identify the basis for the action taken		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(9)	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.1.1 Knowledge of conduct of operations requirements.	Tier	3
	Group #	2 Conduct of Operations
	K/A	G 2.1.1
	Rating	3.8
	Revision	2
Revision Statement: Added the bank source as “#335 GGNS 2011 NRC Exam Q31		

Question: 66

According to EN-OP-115, Conduct of Operations, which of the following activities is authorized for Two-Handed Operation at River Bend Station?

A. Manipulating RCIC Test Return Valves

B. Terminating and preventing injection from High Pressure Core Spray

C. Manual Initiation of Automatic Depressurization System

D. Adjusting reactor power using recirculation FCV controllers

Answer: B

Explanation:

Allowed per EN-OP-115. Acceptable two-handed activities are as follows: Terminate and Prevent Injection from HPCS. This is due to the initiation push button and the injection valve must be manipulated at the same time.(EN-OP-115 page 66, OSP-53 page 41)

Distracters:

A Manipulation of HPCS Test return valves is allowed due to two valves in series and both valves are throttle valves. RCIC has two valves in series also (F059 and F022) but one is a throttle valve and the other is not so there is no need or procedure guidance to operate these valves at the same time. The unprepared applicant may confuse the guidance for the two systems.SOP-35 pages 14, 15, 16)

C There are two divisions of ADS and each division has two arm and depress initiation push buttons. Either division initiation push buttons will open the ADS valves. Due to the close proximity of the two push buttons the unprepared applicant may misunderstand how the initiation push buttons/initiation circuit operates.(SOP-11 page 20)

D RBS operates the recirc flow control system in loop manual mode of operation. Meaning each loop flow control valves are in manual not automatic. Since the recirc loop flows are required by TS to be within 5 or 10 % mismatch operating both FCVs at the same time would allow to maintain the loop flows matched. However this is not allowed. Each FCV controller must be manipulated individually. The unprepared applicant may not have knowledge of this requirement.(SOP-3 page 38)

K/A Match

This question tests the applicants' knowledge of conduct of operations requirements as it applies to acceptable two handed operation.

Technical References:

EN-OP-115 page 66

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

None

Question Source:	**Bank # 335 GGNS 2011 NRC Exam Q# 31**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41.b.10	
Level of Difficulty:	2	

SRO Only Justification: NA

PRA Applicability:

Examination Outline Cross Reference	**Level**	RO
2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.	**Tier**	3
	Group #	1 Conduct of Operations
	K/A	G2.1.4
	Rating	3.3
	Revision	2
Revision Statement: Made part (1) FIVE 12-hour watches vs SEVEN 12-hour watches, and made part (2) test on the eligible watchstations for maintaining proficiency. BOP or ATC only, vs BOP or ATC or Work Control		

Question: 67

In order to maintain license proficiency, an RO must:
(1) stand a MINIMUM of _______ 12 hour watches per calendar quarter
(2) in a combination of ___________ position.

ATC: At-The-Controls Operator
BOP: Balance of Plant Operator
WC: Work Control RO

	(1)	(2)
A.	5	ATC or BOP ONLY
B.	5	ATC, BOP or WC
C.	7	ATC or BOP ONLY
D.	7	ATC, BOP or WC

Answer: A

Explanation:
55.53 Conditions of licenses:
To maintain active status, the licensee shall actively perform the functions of an operator or senior operator on a minimum of seven 8-hour or five 12-hour shifts per calendar quarter.

Additionally, proficiency watches for an RO must consist of ATC or BOP watches. Shifts spent in Work Control do not count for proficiency.

Distracters:
B, Incorrect, shifts spent in Work Control do not count for proficiency.
C. Incorrect, only 5 12 hour watches are required per calendar quarter.

D. Incorrect, only 5 12 hour watches are required per calendar quarter.		
K/A Match This question requires the applicant to have knowledge of individual licensed operator responsibilities related to maintenance of active license status.		
Technical References: 10CFR55.53 Conditions of licenses		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: None		
Question Source:	**Bank #**	2010 RBS Audit
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41.10	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	RO
2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.	**Tier**	3
	Group #	1 Conduct of Operations
	K/A	G 2.1.43
	Rating	4.1
	Revision	1
Revision Statement: Edited stem to "While at 100% power, an unexpected feedwater heater string isolation has occurred." Edit Part 2 to "Per AOP-0007, Loss of Feedwater Heating, which of the following is a potential challenge from this malfunction?"		

Question: 68

While at 100% power, an unexpected feedwater heater string isolation has occurred

(1) What effect will this change have on reactivity?

(2) Per AOP-0007, Loss of Feedwater Heating, which of the following is a potential challenge from this malfunction?

A. (1) Inserts negative reactivity
   (2) Reduces NPSH to the recirculation pumps

B. (1) Inserts negative reactivity
   (2) Challenges fuel integrity

C. (1) Inserts positive reactivity
   (2) Reduces NPSH to the recirculation pumps

D. (1) Inserts positive reactivity
   (2) Challenges fuel integrity

Answer: D
Explanation: AOP-0007 LOSS OF FEEDWATER HEATING gives guidance for a loss of feedwater heating and the resultant addition of cold water to the reactor. At rated power conditions, the resultant cold water injection will be sufficient to provide significant reactor core coolant temperature decrease. This will result in a significant reactor power increase in the lower reactor core region and potential challenges to fuel integrity.AOP-7 page 3

Distracters: A Power will not lower as noted in the explanation above. The reduction in feedwater temperature would improve NPSH to the recirc pumps by reducing recirc pump inlet temperature. Therefore reducing NPSH is not correct, but plausible if applicant does not understand the effects of a feedwater temperature reduction on recirc suction temperature B See above C See above
K/A Match This question tests the applicants ability to determine the effects that changes to feedwater temperature will have on reactor coolant temperature and the resulting changes to core reactivity.
Technical References: AOP-7 pages 3 and 4
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: RLP-HLO-526 Objective 5: Describe how each of the following may be affected by a loss of feedwater heating: (5) a. MCPR limit b. Reactor Power c. RPS d. RC&IS e. Feedwater Nozzles

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41.b.10	
Level of Difficulty:	3	

SRO Only Justification: NA
PRA Applicability:

Examination Outline Cross Reference	Level	RO
2.2.7 Knowledge of the process for conducting special or infrequent tests.	Tier	3
	Group #	2 Equipment Control
	K/A	G2.2.7
	Rating	2.9
	Revision	2
Revision Statement: Modified the stem to characterize the TS surveillance as a maintenance activity requiring special instructions to add plausibility to distractors A, B, and D		

Question: 69

A special test required by Technical Specifications with a 24 month periodicity is in progress. Prior to the performance of the test, specific criteria were established to ensure nuclear safety is maintained during testing.

Which of the following provides guidance for establishing the special instructions used to control the maintenance activity as well as establishing required actions to take if the specified parameters are exceeded?

A. EN-DC-117, POST MODIFICATION TESTING AND SPECIAL INSTRUCTIONS

B. EN-MA-125, TROUBLESHOOTING CONTROL OF MAINTENANCE ACTIVITIES

C. EN-OP-116, INFREQUENTLY PERFORMED TESTS OR EVOLUTIONS

D. EN-WM-107, POST MAINTENANCE TESTING

Answer: C

Explanation:
EN-OP-116 is provided for the performance of special tests or infrequently performed evolutions. The briefing checklist includes the specific criteria used for test termination be covered in the brief as well as the actions to take if specific plant performance criteria has been exceeded. Also it gives examples of when an IPTE is required.(EN-OP-116 page 17, 18, 29)

Distracters:
A Plausible because the applicant if not familiar with EN-OP-116 may select A because the stem states “guidance for establishing the special instructions” and the title is POST MODIFICATION TESTING AND SPECIAL INSTRUCTIONS.
B Plausible because the applicant if not familiar with EN-OP-116 may select D because the stem states “guidance….. used to control the maintenance activity” and the title is TROUBLESHOOTING CONTROL OF MAINTENANCE ACTIVITIES .

D Plausible because the applicant if not familiar with EN-OP-116 may select D because the stem states “guidance….. used to control the maintenance activity”. Also the TS surveillance is referred to as a “test’“ and the title is. POST MAINTENANCE TESTING.		
K/A Match This question determines the applicants knowledge of infrequent test process, specifically the pre job briefing requirements and content.		
Technical References: EN-OP-116		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: None.		
Question Source:	**Bank # April 2010 NRC Q94**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41.10	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.2.23 Ability to track Technical Specification limiting conditions for operations.	Tier	3
	Group #	2 Equipment Control
	K/A	G 2.2.23
	Rating	3.1
	Revision	2

Revision Statement:
Revised answer to ESOMS instead of ESOMS LCO Tracking module to avoid cueing answer. Replaced distractor D Equipment Out Of Service (EOOS) Monitor to improve plausibility/discrimination

Question: 70

Where would the on shift Reactor Operator find a listing of Actual and Potential Technical Specification Limiting Condition for Operations that are in effect?

A. ESOMS

B. Shift Manager Relief Checklist

C. Surveillance Test Scheduling Database

D. Equipment Out Of Service (EOOS) Monitor

Answer: A

Explanation:
Actual and Potential LCO are recorded/tracked using the ESOMS software program, specifically the LCO Tracking Module OSP-0040 Rev 14 page 8 and attachment 4
A licensed Senior Reactor Operator (SRO) will fill out an LCO Status Sheet or computerized version per Attachment 4 whenever the plant enters the action statements of a Technical Specification or Technical Requirement.

ESOMS used as the answer as opposed to the ESOMS LCO Tracking Module to maintain discrimination between the distractors.

Distracters:
B The Shift Manager relief checklist has a section regarding LCO status however it refers back to the LCO tracking log this may confuse the unprepared applicant. (OSP-2 page 9)
C The Surveillance Test Scheduling Database is maintained to ensure TS surveillance requirements are met. This does not track whether are not an LCO has been entered or exited for the period the equipment is inoperable for that testing. The unprepared applicant may confuse the two.(ADM-15 page 6)
D EOOS Monitor is a computer program designed to track out of service equipment

(system, structure or component (SSC)) to quantify the risk of a core damaging event.ADM-0096 RISK MANAGEMENT PROGRAM IMPLEMENTATION AND ON-LINE This is a plausible distractor for an applicant that is not familiar with the ESOMS LCO Tracking Module. He may select D because it is a monitor that tracks out of service equipment, but it tracks risk and not LCO's		
K/A Match This question determines the applicants ability to track TS LCOs through the use of ESOMS.		
Technical References: OSP-40 page 8, ADM-15 page 6, OSP-2 page 9		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-OPS-HLO-OSP40-LO Objective 1: State the purpose of OSP-0040 (1)		
Question Source:	**Bank # 2008 NRC #71**	
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41.b.10	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	Level	RO
2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions.	Tier	3
	Group #	3 Radiation Control
	K/A	G2.3.7
	Rating	3.5
	Revision	1
Revision Statement:		

Question: 71

What type of information would you expect to find on a General Radiation Work Permit (RWP)?

A. An individual's dose margin.

B. Electronic Alarming Dosimeter (EAD) settings

C. High Risk work activity dose rates at Hot Spots.

D. Total department cumulative dose and dose goals.

Answer: B

Explanation:
EAD Settings are determined and provided on each RWP(EN-RP-105 page30, 43

Distracters:
A An individual's dose margin is presented during the login process prior to entry to the RCA and during the RP pre-job brief. The unprepared applicant may confuse the location of this information.EN-RP-105 page 31
C Plausible if applicant confuses General RWP requirements with specific RWP requirements. The General RWP is a low risk RWP and the specific RWP is normally for medium and high risk work activities. EN-OP-105, page 6
D Total department cumulative dose and dose goals are not located on an RWP; however, the cumulative dose and dose estimates for the RWP are. The unprepared applicant may confuse the location of this information.EN-RP-105 page 29

K/A Match
This question determines the applicants ability to comply with radiation work permit requirements during normal conditions by knowing what information / limits are contained on the RWP.

Technical References: EN-RP-105 / 106		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: None		
Question Source:	**Bank # 2008 NRC #72**	
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41.12	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	RO
2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	**Tier**	3
	Group #	3 Radiation Control
	K/A	G 2.3.14
	Rating	3.4
	Revision	3

Revision Statement:
The question originally required detailed procedural knowledge to know which activities would require notification of Radiation Protection personnel. Revised the question to require applicant to recognize which normal evolutions result in an increase in area radiation levels.
Added details to explanation to support changes.
Added justification for negatively phrase question to explanation
Revision 3: Changed stem to remove “not” statement. Changed distractors A and B.

Question: 72

Which of the following evolutions would cause a significant change in local area radiation levels?

A. Alternating CRD flow Control Valves

B. Performing STP-055-0705, Fuel Handling Platform Operability Test

C. Swapping running Reactor Plant Component Cooling Water Pumps.

D. Placing the Hydrogen Water Chemistry System in service at 50% power.

Answer: D

Explanation:

Placing the Hydrogen Water Chemistry System at 50% power causes radiation levels in the turbine building to rise significantly due to N16 production in the reactor. (SOP-123 page 19)

Distracters:
A. Alternating CRD flow control valves will not cause a significant change in local area radiation levels. This is plausible if applicant confuses with operating the recirculation system flow control valves.

B. Performing STP-055-0705 does require movement and testing of all of the equipment on the fuel handling platform, but it does not require the movement of any fuel bundles. Some of the tests are performed using test weights and verifying movement of the fuel handling bridge and its components. A significant change in local

radiation levels is not expected while performing this evolution. C. The RPCCW cools systems that contain radioactive fluids, such as RWCU. However RPCCW is not considered a radioactive system; therefore, swapping RPCCW pumps would not affect local area radiation levels.		
K/A Match This question requires the applicant to have knowledge of normal evolutions that increase area radiation levels.		
Technical References: SOP-90, SOP-123, and STP-055-0705		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-127-LO Objective 12: Analyze the operational impact of the related Operational Experiences associated with the HWC System (12).		
Question Source:	**Bank # 2010 NRC #72**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41.b.12	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="6">2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.</td><td>Tier</td><td>3</td></tr>
<tr><td>Group #</td><td>4 Emergency Procedures/Plan</td></tr>
<tr><td>K/A</td><td>G 2.4.8</td></tr>
<tr><td>Rating</td><td>3.8</td></tr>
<tr><td>Revision</td><td>3</td></tr>
<tr></tr>
<tr><td colspan="3">Revision Statement:
Edited distractors B&C per the above. Also revised explanation by adding: The Area Sump pump control switches would be returned to automatic operation in accordance with the EOPs since “EOP actions are prioritized over AOP actions if the objectives conflict.” This hierarchy between EOP’s and AOP’s is described in the Plant Specific Technical Guidelines (PSTG) as follows:…
Revision 3: Edited distractors A and D. Clarified answer justification.</td></tr>
</table>

Question: 73

A steam line break has occurred. The control switches for the Area Sump pumps are aligned as directed by AOP.

Subsequently, an EOP step is reached which requires operation of all available area sump pumps as necessary to maintain area water levels below Max Normal Operating Values.

Should the Area Sump pump control switches be returned to automatic operation and why or why not?

A. Yes, because chronologically, the EOP action occurred most recently; the most recent chronological action takes precedent when event-specific AOPs are performed in conjunction with EOPs.

B. Yes, because while event-specific AOPs may be performed in conjunction with EOPs, EOP actions are prioritized over AOP actions if the objectives conflict.

C. No, because due to the event-specific nature of AOPs, AOP actions are prioritized over EOP actions if the objectives conflict.

D. No, because chronologically, the AOP action occurred first; the first chronological action takes precedent when event-specific AOPs are performed in conjunction with EOPs.

Answer: B
Explanation:

The Area Sump pump control switches would be returned to automatic operation in accordance with the EOPs since EOP actions are prioritized over AOP actions if the objectives conflict. This hierarchy between EOP's and AOP's is described in the Plant Specific Technical Guidelines (PSTG) as follows: "The EOPs/SAPs address a spectrum of conditions including those more severe as well as those less severe than those considered in plant design bases. While the prescribed actions are appropriate for any initiating event, diagnosis of the event may facilitate plant recovery and enhance the overall response to emergency conditions. Event-specific procedures may thus be developed to supplement procedures developed from the EOPs/SAPs. However, the structure and content of these support procedures must be compatible with the EOPs/SAPs and must not subvert EOP/SAP objectives." EPSTG-2 page B-4-7 EOP-3, Step SC-9, "Isolate all systems discharging into the area except systems required for damage control and systems required to be operated by EOPs." The sump pumps are required per EOP-3, Step SC-6, "Operate all available area sump pumps as necessary to restore and maintain Sec STMT area water levels below Table SC-2 Max Normal Operating Values." If there is a conflict with sump pump operation, step SC-9 allows operation when required to be operated by EOPs.
Distracters: A Although the event based AOP actions were performed first the subsequent EOP actions take priority and override the AOP actions. The unprepared applicant may not understand the prioritization of EOP and AOP actions. C see A D see A
K/A Match This question tests the knowledge of how abnormal operating procedures are used in conjunction with EOP-3.
Technical References: EPSTG-2 page B-4-7
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: RPPT-OPS-HLO-511 Obj 1: State the function of the Emergency Operating Procedures (EOPs).

Question Source:	**Bank # March 2010 Audit**	X

2018 RBS NRC Examination

	#74	
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41.b10	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		
Top 10 Internal Risk Events: 5/10 are Flooding Events		

<table>
<tr><td>Examination Outline Cross Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="6">2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.</td><td>Tier</td><td>3</td></tr>
<tr><td>Group #</td><td>4 Emergency Procedures/Plan</td></tr>
<tr><td>K/A</td><td>G 2.4.22</td></tr>
<tr><td>Rating</td><td>3.6</td></tr>
<tr><td>Revision</td><td>3</td></tr>
<tr><td></td><td></td></tr>
<tr><td colspan="3">Revision Statement:
Referenced the actual step being referred to in stem by adding: “During an ATWS, EOP-1A RPV CONTROL ATWS, Step RCA-3 states “Inhibit ADS” to prevent automatic initiation…. ADS is inhibited to prevent which one of the following?
Revised distractor A from “boron losses” to “cooldown” revise all of the distractors to be consistent by putting the format of “adverse condition and consequence” (as a result of not inhibiting ADS)]
Added details to the explanation for where the information in distractor A is derived from and why distractor A is wrong.
Added details to distractor B to document that voiding is a not a precursor neutron flux oscillation.
Revision 3: Changed distractor A.</td></tr>
</table>

Question: 74

During an ATWS, EOP-1A RPV CONTROL ATWS, Step RCA-3 states “Inhibit ADS” to prevent automatic initiation of the Automatic Depressurization System (ADS).

ADS is inhibited to prevent which one of the following?

A. Erratic indications on Wide Range Reactor Level due to rapid depressurization, which would interfere with effective level power control.

B. Erratic voiding of the core that may result in neutron flux oscillations

C. Low pressure ECCS injection causing a power excursion.

D. Heat addition to the suppression pool causing greater than 110°F Suppression Pool Temperature before boron injection

Answer: C
Explanation: ADS initiation may result in the injection of large amounts of relatively cold, unborated water from low pressure injection systems. With the reactor either critical or shutdown on boron, the positive reactivity addition due to boron dilution and temperature reduction

effected through the injection of cold water may result in a reactor power excursion large enough to cause substantial core damage(EPSTG-2 page B-7-4

Distracters:

A Wide range reactor level instruments are relied on in executing level power control since reactor level is maintained below the narrow range instruments. The distractor is plausible since the wide range level instruments reference legs are susceptible to degassing during rapid depressurization causing erratic indications. However, this is not the reason for inhibiting ADS. Additionally, there is a reference leg backfill system that is designed to prevent erratic indications from the wide range reactor level indicators during rapid depressurization. As stated in STM-51, Nuclear Boiler Instrumentation: “The function of the backfill system is to purge dissolved non-condensible gasses from the reference legs; such that, gas introduced level errors (degassing and notching) will not occur upon depressurization.” An applicant may not be aware of the purpose of the reference leg backfill system.

B Although an ADS blowdown will increase voiding, voiding is not a precursor to neutron flux oscillations. This is the bases for a different step in the Flooding ATWS procedure that identifies inlet subcooling, as a precursor to flux oscillations. The unprepared applicant may confuse the bases steps.EOP-4A step RFQ-3, EPSTG-2 page B-12-31

D While a rising suppression pool temperature will be the result of ADS actuation during ATWS conditions this is not the bases for this action. This is the Minimum Boron Injection Temperature for RBS. The unprepared applicant may confuse the result with the bases.

K/A Match

This question requires knowledge of emergency procedure placing priority on reactivity control during an ATWS over that of pressure control and how the pressure control function effect reactivity.

Technical References:

EPSTG-2 for EOP-1A and EOP-4A

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective: RLP-LO-513 Objective 4: Given an EOP step, identify the basis for the action taken.		
Question Source:	**Bank #**	2003 NRC #98
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	41(b)(5)&(10)	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		
Top 10 Risk Significant Systems: SRV Depressurization		

Examination Outline Cross Reference	Level	RO
2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	Tier	3
	Group #	4 Emergency Procedures/Plan
	K/A	G 2.4.34
	Rating	4.2
	Revision	1
Revision Statement: Corrected distractor C explanation to the match the distractor. Added additional information to the other distractor explanations to explain the basis of the distractor and support the plausibility of each.		

Question: 75

A fire occurs in the Main Control Room that requires evacuation.

(1) What action(s) will an RO perform outside the Main Control Room?
(2) What effect will this action have?

A. (1) Secure both RPS MG sets and place all RPS EPA breakers to OFF
(2) Confirmatory Reactor Scram and full NSSSS Isolation

B. (1) Secure both RPS MG sets and place all RPS EPA breakers to OFF
(2) Initiation of HPCS, LPCS and RCIC

C. (1) Stop all running Reactor Feed Pumps and Condensate Pumps
(2) Initiation of HPCS, LPCS and RCIC

D. (1) Stop all running Reactor Feed Pumps and Condensate Pumps
(2) Confirmatory Reactor Scram and full NSSSS Isolation

Answer: A

Explanation:

This direction provided by the AOP if there is a fire in the main control room(AOP-31 page 9 and 100)

Unit Operator Actions attachment 13 states that if a control room fire is in progress then initiate a confirmatory reactor scram and full, inboard and outboard, NSSS isolation (AOP-31 page 100)

Distracters:

B. part (1) is correct action; however, part (2) is not correct because initiation of HPCS, LPCS and RCIC is not an effect, it is an action that is performed only if a MCR fire is NOT in progress (AOP-31 page 10). This is plausible if applicant confuses the initiation logic power supply for the ECCS systems.

C. Part (1) is a correct action for a MCR fire; however, part (1) is not correct because the action is taken inside the control room. Part (2) could be a possible effect of stopping all running Reactor Feed Pumps and Condensate Pumps; however, initiation of HPCS, LPCS and RCIC is an action that is performed only if a MCR fire is NOT in progress. This distractor is plausible if applicant does not recall that initiation of HPCS, LPCS and RCIC is manual action performed only if a MCR fire is NOT in progress.

D. Part (1) is a correct action for a MCR fire; however, part (1) is not correct because the action is taken inside the control room. Part (2) could be a possible effect of stopping all running Reactor Feed Pumps and Condensate Pumps; however, Confirmatory Reactor Scram and full NSSSS Isolation. However, Confirmatory Reactor Scram and full NSSSS Isolation are carried out by securing both RPS MG sets and placing all RPS EPA breakers to OFF Plausible if applicant confuses part (1) as a correct answer.

K/A Match

This question determines the applicants' knowledge of actions taken outside the main controlroom during a fire and the operational impact of securing RPS MG sets and EPA breakers.

Technical References:

AOP-31 pages 9, 10, 100

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-OPS-AOP31 Objective 5: Given AOP-0031, Shutdown from Outside the Main Control Room and a description of plant conditions, determine the appropriate operator actions. (5)

Question Source:	**Bank #**	
	Modified Bank #	

	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.41b.10	
Level of Difficulty:	3	
SRO Only Justification: NA		
PRA Applicability:		
Top 10 Risk Significant Systems: Reactor Protection System		

MODE SELECTION

1 Time Since Reactor Shutdown

Record current date and time ____________/____________

Record date and time of Reactor shutdown _____________

Determine length of time since Reactor shutdown ______________ Hours

2 Reactor Core Decay Heat from Attachment 6 or Incore Fuels Group

____________________10^6 BTU/HR

Attachment 6 / Incore Fuels Group (Circle one)

MODE SELECTION

NOTE

An Alternate Method is required for each inoperable RHR-SDC mode.

More than one system may be needed to meet Reactor Core Decay Heat requirements; however, credit for each system may be used only once.

3 Compare Reactor Core Decay Heat value from Step **2** of this attachment to the systems heat removal capacities below and determine the Alternate Shutdown Cooling Methods. Record Alternate Shutdown Cooling Methods in Step **6.2.3.2**.

SYSTEM	HEAT REMOVAL CAPACITY (BTU/HR)	CONDITIONS / ASSUMPTIONS
SPC/ADHR	37.67×10^6 40.12×10^6 70.86×10^6	2500 gpm, 83°F Service Water & 2250 gpm, 120°F Rx Coolant 2250 gpm, 140°F Rx Coolant 2250 gpm, 200°F Rx Coolant
FPC Assist	See Attachment 2	2000 gpm, 95°F Service Water
CRD	2.5×10^6	50 gpm, 100°F CRD & 200°F Rx Coolant
Condensate	7.5×10^6	200 gpm, 124°F Condensate & 200°F Rx Coolant
RWCU	See Attachment 3	471 gpm RPCCW** & 248 gpm Rx Coolant
RHR-LPCI	126×10^6	5800 gpm, 95°F Service Water & 5050 gpm, 185°F Rx Coolant
SFC	See Attachment 4	2000 gpm Service Water & 2500 gpm Rx Coolant
MSL Flooding	60×10^6	1758 gpm/20 hrs after shutdown &, 120°F to 170°F Rx Coolant &, 65°F to 100°F Circulating Water

** This pertains to the non-regenerative heat exchanger only.

FPC ASSIST TEMPERATURE DEPENDENT HEAT REMOVAL CAPACITY TABLE

FPC Assist of RHR, Service Water Flow @ 2000GPM							
SW	Heat Removal Capacity (MBtu/hr)						
Tci (°F)	Pool = 95°F	Pool = 100°F	Pool = 105°F	Pool = 110°F	Pool = 120°F	Pool = 130°F	Pool = 140°F
75	11.5	19.17	23.00	26.84	34.50	42.17	49.84
77	9.97	17.64	21.47	25.30	32.97	40.64	48.31
79	8.43	16.10	19.94	23.77	31.44	39.11	46.77
81	6.90	14.57	18.40	22.24	29.90	37.57	45.24
83	5.37	13.04	16.87	20.70	28.37	36.04	43.71
85	3.83	11.50	15.34	19.17	26.84	34.50	42.17
87	2.30	9.97	13.80	17.64	25.30	32.97	40.64
89	.77	8.43	12.27	16.10	23.77	31.44	39.11
91		6.90	10.73	14.57	22.24	29.90	37.57
93		5.37	9.20	13.04	20.70	28.37	36.04
95		3.83	7.67	11.50	19.17	26.84	34.50
97		2.30	6.13	9.97	17.64	25.30	32.97
99		.77	4.60	8.43	16.10	23.77	31.44
101			3.07	6.90	14.57	22.24	29.90
103			1.53	5.37	13.04	20.70	28.37
105				3.83	11.5	19.17	26.84

FPC ASSIST OF RHR, SERVICE WATER FLOW @ 2000GPM						
SW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Pool = 150°F	Pool = 160°F	Pool = 170°F	Pool = 180°F	Pool = 190°F	Pool = 200°F
75	57.51	65.18	72.84	80.51	88.18	95.85
77	55.97	63.64	71.31	78.98	86.65	94.31
79	54.44	62.11	69.78	77.44	85.11	92.78
81	52.91	60.57	68.24	75.91	83.58	91.25
83	51.37	59.04	66.71	74.38	82.04	89.71
85	49.84	57.51	65.18	72.84	80.51	88.18
87	48.31	55.97	63.64	71.31	78.98	86.65
89	46.77	54.44	62.11	69.78	77.44	85.11
91	45.24	52.91	60.57	68.24	75.91	83.58
93	43.71	51.37	59.04	66.71	74.38	82.04
95	42.17	49.84	57.51	65.18	72.84	80.51
97	40.64	48.31	55.97	63.64	71.31	78.98
99	39.11	46.77	54.44	62.11	69.78	77.44
101	37.57	45.24	52.91	60.57	68.24	75.91
103	36.04	43.71	51.37	59.04	66.71	74.38
105	34.50	42.17	49.84	57.51	65.18	72.84

RWCU TEMPERATURE DEPENDENT HEAT REMOVAL CAPACITY TABLES

RWCU						
RPCCW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 90°F	Rx = 100°F	Rx = 110°F	Rx = 120°F	Rx = 130°F	Rx = 140°F
75	1.48	2.47	3.46	4.44	5.43	6.42
77	1.28	2.27	3.26	4.25	5.23	6.22
79	1.09	2.07	3.06	4.05	5.04	6.03
81	0.89	1.88	2.86	3.85	4.84	5.83
83	0.69	1.68	2.67	3.65	4.64	5.63
85	0.49	1.48	2.47	3.46	4.44	5.43
87	0.30	1.28	2.27	3.26	4.25	5.23
89	0.10	1.09	2.07	3.06	4.05	5.04
91		0.89	1.88	2.86	3.85	4.84
93		0.69	1.68	2.67	3.65	4.64
95		0.49	1.48	2.47	3.46	4.44
97		0.30	1.28	2.27	3.26	4.25
99		0.10	1.09	2.07	3.06	4.05
101			0.89	1.88	2.86	3.85
103			0.69	1.68	2.67	3.65
105			0.49	1.48	2.47	3.46

RWCU						
RPCCW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 150°F	Rx = 160°F	Rx = 170°F	Rx = 180°F	Rx = 190°F	Rx = 200°F
75	7.41	8.40	9.38	10.37	11.36	12.35
77	7.21	8.20	9.19	10.17	11.16	12.15
79	7.01	8.00	8.99	9.98	10.96	11.95
81	6.82	7.80	8.79	9.78	10.77	11.75
83	6.62	7.61	8.59	9.58	10.57	11.56
85	6.42	7.41	8.40	9.38	10.37	11.36
87	6.22	7.21	8.20	9.19	10.17	11.16
89	6.03	7.01	8.00	8.99	9.98	10.96
91	5.83	6.82	7.80	8.79	9.78	10.77
93	5.63	6.62	7.61	8.59	9.58	10.57
95	5.43	6.42	7.41	8.40	9.38	10.37
97	5.23	6.22	7.21	8.20	9.19	10.17
99	5.04	6.03	7.01	8.00	8.99	9.98
101	4.84	5.83	6.82	7.80	8.79	9.78
103	4.64	5.63	6.62	7.61	8.59	9.58
105	4.44	5.43	6.42	7.41	8.40	9.38

SFC TEMPERATURE DEPENDENT HEAT REMOVAL CAPACITY TABLES

SSW or RPCCW	SFC Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 90°F	Rx = 100°F	Rx = 110°F	Rx = 120°F	Rx = 130°F	Rx = 140°F
75	6.96	11.61	16.25	20.89	25.54	30.18
77	6.04	10.68	15.32	19.97	24.61	29.25
79	5.11	9.75	14.39	19.04	23.68	28.32
81	4.18	8.82	13.47	18.11	22.75	27.39
83	3.25	7.89	12.54	17.18	21.82	26.47
85	2.32	6.96	11.61	16.25	20.89	25.54
87	1.39	6.04	10.68	15.32	19.97	24.61
89	0.46	5.11	9.75	14.39	19.04	23.68
91		4.18	8.82	13.47	18.11	22.75
93		3.25	7.89	12.54	17.18	21.82
95		2.32	6.96	11.61	16.25	20.89
97		1.39	6.04	10.68	15.32	19.97
99		0.46	5.11	9.75	14.39	19.04
101			4.18	8.82	13.47	18.11
103			3.25	7.89	12.54	17.18
105			2.32	6.96	11.61	16.25

SSW or RPCCW	SFC Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 150°F	Rx = 160°F	Rx = 170°F	Rx = 180°F	Rx = 190°F	Rx = 200°F
75	34.82	39.47	44.11	48.75	53.40	58.04
77	33.90	38.54	43.18	47.82	52.47	57.11
79	32.97	37.61	42.25	46.90	51.54	56.18
81	32.04	36.68	41.32	45.97	50.61	55.25
83	31.11	35.75	40.40	45.04	49.68	54.33
85	30.18	34.82	39.47	44.11	48.75	53.40
87	29.25	33.90	38.54	43.18	47.82	52.47
89	28.32	32.97	37.61	42.25	46.90	51.54
91	27.39	32.04	36.68	41.32	45.97	50.61
93	26.47	31.11	35.75	40.40	45.04	49.68
95	25.54	30.18	34.82	39.47	44.11	48.75
97	24.61	29.25	33.90	38.54	43.18	47.82
99	23.68	28.32	32.97	37.61	42.25	46.90
101	22.75	27.39	32.04	36.68	41.32	45.97
103	21.82	26.47	31.11	35.75	40.40	45.04
105	20.89	25.54	30.18	34.82	39.47	44.11

COMBINED RWCU AND SFC TEMPERATURE DEPENDENT HEAT REMOVAL CAPACITY TABLES

RWCU and SFC						
RPCCW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 90°F	Rx = 100°F	Rx = 110°F	Rx = 120°F	Rx = 130°F	Rx = 140°F
75	8.45	14.08	19.71	25.34	30.97	36.60
77	7.32	12.95	18.58	24.21	29.84	35.47
79	6.19	11.82	17.46	23.09	28.72	34.35
81	5.07	10.70	16.33	21.96	27.59	33.22
83	3.94	9.57	15.20	20.83	26.47	32.10
85	2.82	8.45	14.08	19.71	25.34	30.97
87	1.69	7.32	12.95	18.58	24.21	29.84
89	0.56	6.19	11.82	17.46	23.09	28.72
91		5.07	10.70	16.33	21.96	27.59
93		3.94	9.57	15.20	20.83	26.47
95		2.82	8.45	14.08	19.71	25.34
97		1.69	7.32	12.95	18.58	24.21
99		0.56	6.19	11.82	17.46	23.09
101			5.07	10.70	16.33	21.96
103			3.94	9.57	15.20	20.83
105			2.82	8.45	14.08	19.71

RWCU and SFC						
RPCCW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 150°F	Rx = 160°F	Rx = 170°F	Rx = 180°F	Rx = 190°F	Rx = 200°F
75	42.23	47.86	53.49	59.12	64.76	70.39
77	41.11	46.74	52.37	58.00	63.63	69.26
79	39.98	45.61	51.24	56.87	62.50	68.13
81	38.85	44.48	50.12	55.75	61.38	67.01
83	37.73	43.36	48.99	54.62	60.25	65.88
85	36.60	42.23	47.86	53.49	59.12	64.76
87	35.47	41.11	46.74	52.37	58.00	63.63
89	34.35	39.98	45.61	51.24	56.87	62.50
91	33.22	38.85	44.48	50.12	55.75	61.38
93	32.10	37.73	43.36	48.99	54.62	60.25
95	30.97	36.60	42.23	47.86	53.49	59.12
97	29.84	35.47	41.11	46.74	52.37	58.00
99	28.72	34.35	39.98	45.61	51.24	56.87
101	27.59	33.22	38.85	44.48	50.12	55.75
103	26.47	32.10	37.73	43.36	48.99	54.62
105	25.34	30.97	36.60	42.23	47.86	53.49

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date:	Facility/Unit: **River Bend Station**
Region: I ☐ II ☐ III ☐ IV ■	Reactor Type: W ☐ CE ☐ BW ☐ GE ■
Start Time:	Finish Time:

Instructions

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

Applicant Certification

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

____
Applicant's Signature

Results		
Examination Value	__________	Points
Applicant's Score	__________	Points
Applicant's Grade	__________	Percent

Examination Outline Cross Reference	Level	SRO
209001 A2.09 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: A2.09 Low suppression pool level.	Tier	2
	Group #	1
	K/A	209001 A2.09
	Rating	3.3
	Revision	3
Revision Statement: Rev 2, Changed suppression pool and CST level to 13 feet. Revision 3: Added (3) to stem and answers. Requires applicant to determine tech spec required action and completion time to mitigate plant conditions.		

Question: 76

The plant is in Mode 5 with a refuel outage in progress.

The upper containment fuel pool gate closed.

RHR A, RHR B, and RHR C are all INOPERABLE due to maintenance.

The suppression pool has been inadvertently drained to 13 feet.

The CST level is 13 feet.

(1) What is the impact on the operability of LPCS?

(2) With no further corrective action, when is the latest time that action must be initiated to suspend OPDRVs per TS 3.5.2?

	(1)	(2)
A.	LPCS is INOPERABLE	Immediately
B.	LPCS is INOPERABLE	4 hours
C.	LPCS is OPERABLE	Immediately
D.	LPCS is OPERABLE	4 hours

Answer: B
Explanation: Question requires applicant to recognize that LPCS is INOPERABLE in part (1), and

then apply TS 3.5.2, which implicitly requires applicant to recognize that HPCS is still OPERABLE, therefore TS 3.5.2.A initially applies with a 4 hour completion time to restore LPCS, after which TS 3.5.2.B applies to Immediately initiate action to suspend OPDRVs.

LPCS is INOPERABLE because it can only take suction from the suppression pool and the level is below the minimum water level.

HPCS is OPERABLE because it can take suction from the CST and the CST water level is sufficient.

Per Tech Spec Bases B3.5.2, the minimum water level of 13 ft 3 inches required for the suppression pool is periodically verified to ensure that the suppression pool will provide adequate net positive suction head (NPSH) for the ECCS pumps, recirculation volume, and vortex prevention. With the suppression pool water level less than the required limit, all ECCS injection/spray subsystems are inoperable unless they are aligned to an OPERABLE CST.

When the suppression pool level is < 13 ft 3 inches, the HPCS System is considered OPERABLE only if it can take suction from the CST and the CST water level is sufficient to provide the required NPSH for the HPCS pump. Therefore, a verification that either the suppression pool water level is ≥ 13 ft 3 inches or the HPCS System is aligned to take suction from the CST and the CST contains ≥ 125,000 gallons of water, equivalent to 11 ft 1 inch, ensures that the HPCS System can supply makeup water to the RPV.

Per TS 3.5.2, Condition A One required ECCS injection/spray subsystem inoperable. A.1 Restore required ECCS injection/spray subsystem to OPERABLE status in 4 hours.

Distracters:

A. Plausible if applicant recognizes that LPCS is INOPERABLE in part 1, but erroneously believes HPCS is INOPERABLE also, requiring an immediate action to suspend OPDRVs
B. Correct
C. Plausible if applicant mistakes HPCS operability requirements for LPCS, and misapplies TS 3.5.2 guidance.
D. Plausible if applicant mistakes HPCS operability requirements for LPCS, but properly applies TS 3.5.2 guidance.

K/A Match

The applicant must first analyze the given conditions and determine the low suppression pool level. Based on the low suppression pool level the applicant must then apply tech spec requirements to determine the operability status of HPCS and LPCS using the requirements defined in tech spec bases. The applicant must also use the operability status to determine the required tech spec condition, required action, and completion time.

Technical References:

Tech Spec Bases B3.5.2, ECCS - Shutdown

Handouts to be provided to the Applicants during exam:

Tech spec 3.5.2 with above the line information grayed out. Also surveillance requirements are removed and grayed out on the bottom of 3.5.2 second page.

Learning Objective:

RLP-STM-0205-LO, Objective (12): Identify the Technical Specifications, Technical Requirements Manual, and/or Bases requirements for the Low Pressure Core Spray (LPCS) System. (12)

Question Source:	**Bank # NRC 2010 #82**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(2)	
Level of Difficulty:	3	

SRO Only Justification:

Applicant must have knowledge of the tech spec bases requirements to determine operability of HPCS and LPCS and determine required tech spec completion time for required action.

PRA Applicability:

Examination Outline Cross Reference	Level	SRO
264000 Emergency Generators (Diesel/Jet) 2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.	Tier	2
	Group #	1
	K/A	264000 2.4.20
	Rating	4.3
	Revision	4

Revision Statement:
Rev 3: Changed stem of question to include station blackout and RCIC tagged out for maintenance. This requires the EDG to be manually started and changes correct answer to B.
Revision 4: Changed part 1 and added to stem HPCS DG failed to automatically start. Removed from stem that RCIC is tagged out for maintenance. Added RPV level is -100 inches and slowly lowering.

Question: 77

EOP-1, RPV Control is entered due to LOCA.

A station blackout is in progress.

HPCS Diesel Generator (DG) failed to automatically start.

RPV level is -100 inches and slowly lowering.

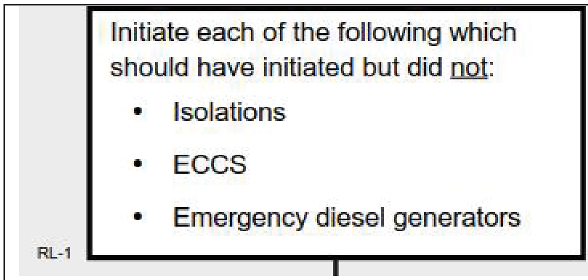
Per EOP-1, the CRS should direct ___(1)___.

RPV water level band should be maintained ___(2)___.

	(1)	(2)
A.	manual start of the HPCS DG immediately	10 to 51 inches
B.	manual start of the HPCS DG immediately	-20 to 51 inches
C.	Emergency Depressurization of the RPV when below TAF	10 to 51 inches
D.	Emergency Depressurization of the RPV when below TAF	-20 to 51 inches

Answer: B

Explanation:



Per EPSTG*0002, EOP-1 Step RL-1, The third bullet “Emergency diesel generators” applies because electrical power supplied by a diesel generator may be required to operate RPV injection systems under emergency conditions. If system operation does not require power to be supplied by an emergency diesel generator, its initiation should be verified when the systems which require the diesel generator are placed in service.

The HPCS Diesel generator is required to operate HPCS due to the loss of offsite power.

Per EOP-1 (Step RL-3) and OSP-53, RPV level band should be -20 to 51 inches for NON-ATWS (expanded).

Per OPS-53, Attachment 1A 2.5, in non-ATWS conditions, the preferred level band is 10 to 51 inches. Under certain plant conditions described in the EOP bases, a wider or lower level band may have been prescribed. Examples include but are not limited to: If the MSIV’S are closed or in anticipation of their closing, OR if a non throttleable injection source is being used to control RPV level, then widen the band to -20” to 51”.

Distracters:

(1) Per EOP-1, RPV Control, Step ALC-4, when RPV level drops to -162 inches (Top of Active Fuel, TAF), and when RPV level cannot be restored and maintained above -187 inches (ALC-8), Emergency Depressurization is required (ALC-11).

(2) 10 to 51 inches is plausible if applicant confuses the normal level band with LPCS control band. LPCS is throttleable; therefore, the normal band 10 to 51 inches is appropriate. HPCS is not throttleable; therefore the expanded band should be used.

K/A Match

Applicant must have knowledge of EOP-1, Step RL-1 and the bases for that step. The applicant must also have knowledge of the note for step RL-3 and apply the guidance from OSP-53 to select the correct level band.

Technical References:

EOP-1, RPV Control EPSTG*0002 OSP-53		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-HLO-512, Objective 5: Given an EOP step identify the basis for the action taken.		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: Applicant must have knowledge of EOP basis documents and station expectations for HPCS Diesel operation. Applicant must be able to assess plant conditions and then determine the RPV level band to mitigate the event.		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
218000 Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.05 Loss of A.C. or D.C. power to ADS valves	Tier	2
	Group #	1
	K/A	218000 A2.05
	Rating	3.6
	Revision	2
Revision Statement: Added reference to December 2014 NRC exam #41. Table 3.8.9-1 is included in TS bases which will not be part of the student handout.		

Question: 78

A short has resulted in the loss of ENB-PNL02A.

Plant conditions require Emergency Depressurization.

To accomplish the Emergency Depressurization, the SRVs should be opened at ___(1)___.

Per technical Specifications, the applicable distribution system must be restored in ___(2)___ hours.

	(1)	(2)
A.	H13-P601	8 hours
B.	H13-P631	8 hours.
C.	H13-P601	2 hours
D.	H13-P631	2 hours

Answer: D
Explanation: ENB PNL02A supplies DC power to the Div 1 SRV opening solenoids. ENB-PNL02B supplies DC power to the Div 2 SRV opening solenoids. With only the loss of PNL02A the Div 2 solenoids are not affected and can be opened normally(energize to open). The normal control switches for the Div 2 SRVs are located on H13-P631(a back panel). The Allowed Outage Time for one DC electrical power distribution subsystem is 2 hours per

TS 3.8.9.C. the distribution subsystems are defined in TS bases 3.8.9 Table B 3.8.9-1.
Distracters: (1) Plausible if applicant confuses SRV operation with normal location for SRV operation and does not recognize failure due to DC bus loss. (2) 8 hours is plausible if applicant confuses ENB-PNL02A bus loss with AC subsystem loss and applies TS 3.8.9 Condition A.
K/A Match The applicant will need to know the impact of the DC power loss on the operation of the ADS system and the TS allowed outage time for loss of a subsystem to mitigate the loss.
Technical References: TS 3.8.9, TS bases 3.8.9, STM-202 rev2
Handouts to be provided to the Applicants during exam: TS 3.8.9
Learning Objective: RLP-STM-202 Objectives: Describe the interrelationships between ADS and the following systems: (6) Electrical Distribution Identify the ADS operability requirements of Technical Specifications Operability (8)

Question Source:	**Bank #**	
	Modified Bank #	Dec 2014 NRC #41 Changed to SRO
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(2)	

Level of Difficulty:	3	
SRO Only Justification: The applicant must interpret the bus loss and the affect that has on the ability to Emergency Depressurize the reactor. The applicant must also determine the correct tech spec action statement and completion time to mitigate the bus loss.		
PRA Applicability:		
Top 10 Risk Significant Systems: SRV Depressurization		

Original question:

November 2010 River Bend Station
Initial License AUDIT Examination
Reactor Operator

QUESTION 40
The plant is operating in the Emergency Operating Procedures following a significant transient. During the transient, a short resulted in the loss of ENB-PNL02A.
Plant conditions require Emergency Depressurization per the Emergency Operating Procedures.
Which of the following represents the method that should be used to accomplish Emergency Depressurization?

A. At H13-P601, open 7 ADS/SRVs.
B. At H13-P631, open 7 ADS/SRVs.
C. Arm and depress the Division 1 ADS Manual Initiate pushbuttons.
D. Alternate depressurization methods listed in the EOPs should be utilized due to SRV failure.

Examination Outline Cross Reference	**Level**	SRO
263000 D.C. Electrical Distribution 2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	**Tier**	2
	Group #	1
	K/A	263000 G 2.2.25
	Rating	4.2
	Revision	2
Revision Statement: Rev 2, Rephrased the stem to provide clarification. Further explained ELAP requirements in explanation.		

Question: 79

Per Technical Specification Bases 3.8.4, DC Sources – Operating, Applicable Safety Analysis, DC Sources are required to be OPERABLE to satisfy the design basis accident analysis for which event(s)?

A. an Extended Loss of all AC Power (ELAP) ONLY.

B. an Extended Loss of all AC Power (ELAP) AND a worst case single failure.

C. a loss of all offsite AC power OR of all onsite AC power ONLY.

D. a loss of all offsite AC power OR of all onsite AC power; AND a worst case single failure.

Answer: D

Explanation:

Per Tech Spec bases for 3.8.4, DC Sources – Operating, the OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining DC sources OPERABLE during accident conditions in the event of: An assumed loss of all offsite AC power or of all onsite AC power; and a worst case single failure.

Distracters:

AOP-65, Extended Loss of AC Power (ELAP), purpose is to provide instructions in event of loss of all offsite and onsite AC power, except for 120VAC supplied by inverters powered from the DC distribution system, lasting or expected to last greater than one hour. This procedure assumes the emergency DC distribution systems and associated batteries and inverters are available. Also, this procedure implements strategies to respond to a Beyond-Design-Basis External Event. Implementation of these strategies may warrant application of 10CFR50.54(x),(y) in order to respond to plant conditions. Technical specifications do not implement strategies to respond to beyond design basis accidents.

K/A Match

The applicant must have the required knowledge of Tech Spec bases 3.8.4, DC Sources – Operating, to select the correct answer.

Technical References:

Tech Spec bases 3.8.4, DC Sources – Operating
AOP-65, Extended Loss of AC Power (ELAP)

Handouts to be provided to the Applicants during exam:
NONE

Learning Objective:

RLP-STM-0305-LO DC, Objective 7: Identify the Technical Specifications, Technical Requirements Manual, and/or Bases requirements for the DC Electrical Distribution System. (7)

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	43(b)(5)	
Level of Difficulty:	3	

SRO Only Justification:

Applicant must have knowledge of Tech Spec bases 3.8.4, DC Sources – Operating.

PRA Applicability:

Top 10 Internal Risk Events: LOSP
Top 10 Risk Significant Systems: 230 kV AC Power

Examination Outline Cross Reference	Level	SRO
510000 Ability to (a) predict the impacts of the following on the SWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: A2.01 Loss of SWS pump	Tier	2
	Group #	1
	K/A	510000 A2.01
	Rating	3.4
	Revision	1
Revision Statement:		

Question: 80

While operating in Mode 1, Normal Service Water (NSW) pumps SWP-P7A and SWP-P7B were running.

SWP-P7B trips on over-current and both divisions of Standby Service Water have initiated.

Based on this, SWP-P7C will start when ____(1)_____, and CRS will enter AOP53 (INITIATION OF STANDBY SERVICE WATER WITH NORMAL SERVICE WATER RUNNING), and direct from subsequent actions to ___(2)_____.

	(1)	(2)
A.	header pressure decreases ≤97 psig	isolate SSW Division II and place redundant SSW Division I in service
B.	when SWP-P7B trips	isolate SSW Division II and place redundant SSW Division I in service
C.	header pressure decreases ≤97 psig	depress the TRIP/STOP Pushbutton for one of the two running NSW pumps
D.	when SWP-P7B trips	depress the TRIP/STOP Pushbutton for one of the two running NSW pumps

Answer: D

Explanation:

The Service Water System (SWP) is comprised of two interconnected subsystems: the Normal Service Water (NSW) sub-system and the Standby Service Water (SSW) sub-system. Essentially, one portion of the Service Water System is non-safety-related; the other is safety-related. Under normal conditions, the Normal SWP pumps provide cooling to both safety and non-safety related loads. Under emergency conditions the two portions of the system will be isolated from each other. The Normal Service Water pumps will serve the non-safety-related loads while the Standby Service Water pumps will serve the safety-related loads. In this condition AOP-53 directs to reduce to one normal service water pump running to prevent pumping the closed loop NSW water into the SSW system. This could drain the surge tank for NSW pumps. The standby NSW pump will auto start on the trip of a running pump or discharge pressure of less than 97 psig. For the conditions stated in the stem standby pump will start on the running pump trip before discharge pressure lowers to 97 psig(normal pressure for NSW is 133 psig)

Distracters:

A (1) The standby NSW pump will start on a low pressure of 97psig however that is not the signal that will start the pump. The pump will start when the running pump trips and before the discharge lower to less than 97psig. (2) This would be appropriate if the SSW system did not initiate and the NSW system was still suppling both NSW and SSW. However the guidance in AOP-53 requires that one NSW pump be secured.

B (1) This part is correct for the conditions given. See explanation above. (2) This part is incorrect as stated in A above.

C (1) This part is incorrect as stated in A above. (2) This part is correct for the conditions given. See explanation above.

K/A Match

This question requires the applicant to understand the impact that a loss of one NSW pump will have on the whole system and direct proper procedure actions to mitigate the abnormal condition.

Technical References:

AOP-9, LOSS OF NORMAL SERVICE WATER,
AOP_53, INITIATION OF STANDBY SERVICE WATER WITH NORMAL SERVICE WATER RUNNING,
STM-118, 5ERVICE WATERSYSTEMS

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective: RLP-STM-118 objective 2 & 9: Describe the automatic functions and interlocks of the following Normal Service Water System components: (2) Normal Service Water Pumps Describe the integrated plant response and associated operator actions following: (9) Loss of Normal Service Water		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: The SRO will assess plant conditions, enter AOP-53 and direct the correct procedure section (5.2 versus 5.1) to mitigate the given abnormal system alignment.		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
223001 Primary Containment System and Auxiliaries A2. 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	**Tier**	2
	Group #	2
	K/A	223001 A2.06
	Rating	4.1
	Revision	1
Revision Statement: Rev 1, changed stem to match verbiage in the EOP.		

Question: 81

While implementing EOP-2, Primary Containment Control, as primary containment pressure rises during an event, an Emergency Depressurization is required when the safe zone of the ___(1)___ graph can no longer be ___(2)___.

HCTL - Heat Capacity Temperature Limit

PSP - Pressure Suppression Pressure

	(1)	(2)
A.	HCTL	maintained
B.	HCTL	restored and maintained
C.	PSP	maintained
D.	PSP	restored and maintained

Answer: D

Explanation:

Per EOP-2, Primary Containment Control, step CP-4, before CTMT pressure reaches the Pressure Suppression Pressure (PSP) (Fig. 4), enter EOP-1. Per step CP-7, when CTMT pressure cannot be restored and maintained below the PSP (Fig. 4), then Emergency Depressurization is required. The applicant must recall this decision point from memory and apply to the correct figure.

Distracters:

Per EOP-2, Primary Containment Control, step SPT-7, when SP temperature and RPV pressure cannot be maintained below the HCTL (Fig. 2), then Emergency Depressurization is required.

A. Plausible because HCTL is also listed as a concern in EOP-2 and it must be maintained below to prevent emergency depressurization.
B. Plausible because correct action but wrong graph.
C. Plausible because correct graph, but the HCTL must be maintained below.

K/A Match

This question requires the applicant to understand the impact of high containment pressure and apply that to the decision point in EOP-2. The applicant must understand the decision point and apply the correct graphical application to ensure the correct mitigating step is taken.

Technical References:

EOP-2, Primary Containment Control

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RPPT-OPS-HLO-517, Objective 2: Given the EOPs/SAPs and plant conditions, discuss the bases for each curve and variable and determine if the plant is in the Safe or Unsafe region of the curve or within the limits of the variable.
Heat Capacity Temperature Limit (HCTL)
Pressure Suppression Pressure (PSP)

Question Source:	**Bank # 2016 Audit #92**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive /	X

	Analysis	
10CFR Part 55 Content:	43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: The applicant must have knowledge of diagnostic steps and decision points in EOP-2 that involve transitions to emergency contingency procedures such as emergency depressurization.		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	SRO
241000 Reactor/Turbine Pressure Regulating System 2.1.32 Ability to explain and apply system limits and precautions.	**Tier**	2
	Group #	1
	K/A	241000 2.1.32
	Rating	4.0
	Revision	2
Revision Statement:		

Question: 82

During an ATWS, when pressure control has been stabilized and SRVs are being used to augment bypass valve pressure control, the CRS directs pressure set to be lowered and the assigned pressure band should be ___(1)___ psig.

The reason for the lower limit to the pressure band is to ___(2)___ the plant cooldown.

	(1)	(2)
A.	500 to 1090	minimize
B.	800 to 1090	minimize
C.	500 to 1090	maximize
D.	800 to 1090	maximize

Answer: B

Explanation:

Per OSP-53, Attachment 1B, Post Scram Pressure Control Strategies, 2.5 if desired, after the initial pressure stabilization, the pressure band may be expanded to 800 psig to 1090 psig. If this expanded pressure band is utilized, and SRVs are being used to augment bypass valve pressure control, pressure set may have to be lowered to ensure bypass valves remain full open and energy is not unnecessarily being deposited in the suppression pool.

Per EPSTG*0002, a "Nominal" RPV pressure band of 800 to 1090 psig minimizes the potential for RPV cooldown to raise reactor power and RPV pressure from exceeding the scram setpoint.

Distracters:

(1) 500 to 1090 is plausible if applicant confuses pressure band for non-ATWS pressure control.

(2) Maximizing cooldown is plausible if applicant confuses strategy with non-ATWS pressure control. Per OSP-53, Attachment 1B, 1.9.1 2. after initial transient, when both level and pressure have been stabilized, the pressure control leg of the EOPs directs that the RPV be depressurized within a 100 degree per hour cooldown rate. The target cool down rate should be set at less than 85 degrees per hour under normal SCRAM conditions which aligns with the 500-1090 PSIG nominal band.

K/A Match

Applicant must interpret plant conditions and apply the correct reactor pressure control strategy. The applicant must also explain the reason for the lower limit of the pressure control band.

Technical References:

OSP-53, EMERGENCY AND TRANSIENT RESPONSE SUPPORT PROCEDURE
EPSTG*002
EOP-1A, RPV Control – ATWS, Step RPA-5

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-LO-513, Objective 6: Given flowcharts for EOP-1A, RPV Control – ATWS, and EOP-4A, Contingencies – ATWS, and plant conditions; apply the appropriate OSP-0053, Emergency and Transient Response Support Procedure, strategy.

Question Source:	**Bank #**	
(note changes and attach parent)	**Modified Bank #**	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X

10CFR Part 55 Content:	43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: Applicant is required to select an EOP decision point for pressure control strategy and provide reason for strategy.		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	SRO
286000 Fire Protection System 2.1.28 Knowledge of the purpose and function of major system components and controls.	**Tier**	2
	Group #	2
	K/A	286000 G2.1.28
	Rating	4.1
	Revision	0
Revision Statement:		

Question: 83

While implementing Alternate Level Control using table L-2 (Alternate Injection Subsystems), per EOP-1, RPV Control, which EOP enclosure is used to inject into the RPV via Standby Service Water and RHR B ONLY?

A. Enclosure 6, INJECTION INTO RPV WITH CONDENSATE TRANSFER

B. Enclosure 7, INJECTION INTO RPV WITH FIRE SYSTEM

C. Enclosure 8, INJECTION INTO RPV WITH SLC TEST TANK

D. Enclosure 35, SUPPRESSION POOL CLEANUP/ALTERNATE DECAY HEAT REMOVAL OPERATION

Answer: B
Explanation: ENCLOSURE 7, INJECTION INTO RPV WITH FIRE SYSTEM, is used to provide instruction for injecting into the RPV with Fire Protection Water System via Standby Service Water and RHR B. Per EOP-1, RPV Control, Alternate Level Control step ALC-3, restore and maintain RPV level above -162 in. with any Table L-1, Preferred Injection Systems. Use Table L-2, Alternate Injection Subsystems, if necessary. Table L-2, Alternate Injection Subsystems, lists condensate transfer (ENCL 6); Fire Water System (ENCL 7); SLC test tank (ENCL 8); and SPC/ADHR (ENCL 35).

Distracters:

ENCLOSURE 6, INJECTION INTO RPV WITH CONDENSATE TRANSFER, is used to provide instructions for injecting into the RPV with Condensate Transfer pumps. This is plausible because it uses RHR A, RHR B, and RHR C.

ENCLOSURE 8, INJECTION INTO RPV WITH SLC TEST TANK, is used to provide instruction for injecting water into the RPV from the SLC test tank with the SLC pumps. This is plausible because applicant may remember flow path uses Condensate makeup and confuse with Enclosure 6 with can be injected via RHR A, RHR B, and RHR C.

ENCLOSURE 35, SUPPRESSION POOL CLEANUP/ALTERNATE DECAY HEAT REMOVAL OPERATION, is used to provide instructions for using the Suppression Pool Cleanup/Alternate Decay Heat Removal System to pump water from the suppression pool into the RPV. This is plausible because this enclosure uses RHR C to inject into the RPV.

K/A Match

Applicant must have basic system understanding of fire water system flow paths and knowledge of applicable enclosures required to provide makeup water to the RPV.

Technical References:

EOP-1, RPV Control
EOP-5, Emergency Operating and Severe Accident Procedures Enclosures

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-HLO-516-ILO Objective 1: Given the applicable Enclosure, and Flowcharts, determine the purpose, method of implementation, and resulting system response for each enclosure (1).

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	

	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: Applicant must demonstrate knowledge of when to implement attachments, including how to coordinate these items with other procedure steps. Based on given information of which systems are available, the applicant must select the EOP enclosure to install to provide injection to the RPV.		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
295003 Partial or Complete Loss of A.C. Power AA2. Ability to determine and/or interpret the following as they apply to partial or complete loss of A.C. power: AA2.04 system lineups	Tier	1
	Group #	1
	K/A	295003 AA2.04
	Rating	3.7
	Revision	4
Revision Statement: Rev 4: Changed stem to Division 2 DG is tagged out for maintenance.		

Question: 84

A loss of RSS-1 and 2 has occurred. Division 1 and 3 diesel generators are operating normally and Division 2 diesel generator is tagged out for maintenance.

The CRS has entered EOP-1, RPV Control and AOP-1, Reactor Scram.

What additional procedure transition is required for proper system lineup based on priority and availability?

A. Install enclosure 20 (DEFEATING DRYWELL COOLING ISOLATION INTERLOCKS) and per AOP-4 restore DRS-UC1A/C/E, Drywell Unit Coolers

B. Per SOP-115, Service Water Cooling, Open / Verify Open SWP-AOV599, STBY CLG TWR INLET, STATION BLACKOUT RETURN TO STBY COOLING TOWER

C. Per AOP-4 (Loss of Offsite Power), pull breaker control power fuses for Division 2 ECCS Pump Breakers.

D. Per SOP-37, Fire Protection Water System Operating Procedure, Close valve FPW-V252, PS-1 ISOLATION VALVE, RCIC room sprinkler system supply.

Answer: A
Explanation: Per AOP-4, Supplemental Actions, if a LOCA does not exist then Drywell Cooling should be manually restored. Per OSP-53, EMERGENCY AND TRANSIENT RESPONSE SUPPORT PROCEDURE, Attachment 35, Loss of Offsite Power, the priority with time critical actions is to Install enclosure 20 and start drywell cooling.

Distracters:
B –. This is incorrect but a plausible step because it is performed during a station blackout if Division 3 diesel generator is operating (which it is) in order to lineup the Div 3 DG to supply power to ENS-SWG1A

C – This is plausible because it is a task per OSP-53, EMERGENCY AND TRANSIENT RESPONSE SUPPORT PROCEDURE, Attachment 2B, Initiating Division 2 Standby Diesel Generator. PRIOR TO REENERGIZING THE DIV 2 SWITCHGEAR, THE DIV 2 ECCS PUMP BREAKERS CONTROL POWER FUSES SHOULD BE PULLED UNTIL THE SYSTEM HAS BEEN VERIFIED FILLED AND VENTED.

D - This is incorrect but a plausible step because it is performed during a station blackout to prevent flooding in the RCIC room.

K/A Match

The applicant must interpret the indications for a partial loss of AC power and apply that to plant system lineups to determine the required action to restore the priority system.

Technical References:

AOP-4 Loss of Offsite Power REV61, AOP-50 REV60,
AOP-50, Station Blackout
OSP-53, EMERGENCY AND TRANSIENT RESPONSE SUPPORT PROCEDURE
SOP-37, Fire Protection Water System Operating Procedure
SOP-115, Service Water Cooling

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RPPT-HLO-523, Objective 5: Describe any immediate and subsequent operator actions required by AOP-0004 (5)

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	

	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: The applicant must determine the required procedure transition and priority for system restoration.		
PRA Applicability:		
Top 10 Internal Risk Events: LOSP & Reactor Trip/Turbine Trip		

Examination Outline Cross Reference	Level	SRO
295019 Partial or Complete Loss of Instrument Air 2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	Tier	1
	Group #	1
	K/A	295018 G 2.4.8
	Rating	4.5
	Revision	2
Revision Statement:		

Question: 85

The plant was operating at 100% and an Instrument Air leak was reported in containment.

Instrument Air header pressure is 40 psig and slowly lowering.

AOP-8, Loss of Instrument Air, was entered.

Reactor was scrammed and all other subsequent actions for a loss of Instrument Air have been completed.

(1) What **FINAL** RPV reactor water level band should be directed?

(2) Pressure control should be maintained using ___(2)___

	(1)	(2)
A.	-20 to 51 inches	SRVs and RCIC ONLY
B.	-20 to 51 inches	steam drains, SRVs, and RCIC
C.	45 to 100 inches	SRVs and RCIC ONLY
D.	45 to 100 inches	steam drains, SRVs, and RCIC

Answer: B
Explanation:

Per EOP-1, RPV Control, Step RL-3, The expanded RPV level band is -20 to 51 inches.

Per EOP-1, Step RP-2, the crew should stabilize PRV pressure below 1090 psig with Main steam line drains and table P-1 Alternate RPV Pressure Control Systems if necessary. SRVs and RCIC are listed as P-1 systems.

Per OSP-53, Attachment 1A, Post Scram Level Control Strategies
In non-ATWS conditions, the preferred level band is 10 to 51 inches. Under certain plant conditions described in the EOP bases, a wider or lower level band may have been prescribed. Examples include but are not limited to: If the MSIV'S are closed or in anticipation of their closing, OR if a non throttleable injection source is being used to control RPV level, then widen the band to -20" to 51".

Per OSP-53, Attachment 1B Post Scram Pressure Control Strategies, Pressure Control Method,
1.2 Post-Scram Pressure Control for an MSIV Isolation.
1.2.1. IF only the inboard MSIVs close due to a loss of air to containment, THEN perform the following:
1. Take manual control of the inboard MSIVs by taking the control switch of each valve to CLOSE.
2. Utilize available steam drains to control pressure.
3. IF required, THEN augment pressure control with SRVs. Each SRV cycle should be closely coordinated with the ATC operator.

Distracters:

(1) Per EOP-1, Step RL-3, Natural Circulation level band is 45 to 100 inches. Plausible if applicant confuses Loss of Instrument Air and Loss of CCP AOP subsequent actions.

(2) Plausible if applicant confused signal to close MSIVs with isolation signal and not due to the loss of instrument air. Main steam line drains would not be available if isolation signal caused the MSIVs and drains to close. The loss of instrument air only closes the MSIVs.

K/A Match

Applicant must have knowledge of subsequent actions required for complete loss of Instrument Air abnormal operating procedure and in conjunction, must be able to apply EOP-1 steps to select the appropriate RPV level band and method of pressure control.

Technical References:

AOP-8, Loss of Instrument Air

EPSTG*0002, Appendix A, River Bend Station Emergency Operating and Severe Accident Procedures and Variable and Curve Bases. EOP-1, RPV Control OSP-53, Emergency and Transient Response Support Procedure		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-OPS-AOP08, Objective 13: Given a set of plant conditions, predict the effect that a loss or malfunction of the Instrument Air System will have on the following: (13) a. Containment Air b. Systems having pneumatic valves and controls		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: Applicant must assess plant conditions for the complete loss of Instrument air and have the knowledge of the subsequent actions taken. Then the applicant must coordinate the AOP-8 entry and EOP-1 entry to direct the correct RPV level band and method of pressure control.		
PRA Applicability:		
Top 10 Internal Risk Events: Reactor Trip/Turbine Trip		

Examination Outline Cross Reference	**Level**	SRO
295023 Refueling Accidents 2.4.41 Knowledge of the emergency action level thresholds and classifications.	**Tier**	1
	Group #	1
	K/A	295023 / 2.4.41
	Rating	4.6
	Revision	1
Revision Statement: Rev 1: Added to stem statement that all other building dose rates are reading normal. Added Attachment 1 to list of provided references.		

Question: 86

During refueling, a fuel assembly is dropped in the spent fuel pool. All workers are evacuated and no one exceeds administrative radiation exposure limits.

Area dose rates on RMS-RE192, South Operating Floor in the Fuel Building continue to rise from a normal reading of 1.563 mR/hr to 1.825E+3 mR/hr.

All other building dose rates are reading normal levels.

(1) What is the classification for this event?

(2) When is the NRC required to be notified?

	(1)	(2)
A.	Notification of Unusual Event	Within 15 minutes of the declaration
B.	Notification of Unusual Event	Immediately after notifying state and local authorities and not later than one hour after declaration
C.	Alert	Within 15 minutes of the declaration
D.	Alert	Immediately after notifying state and local authorities and not later than one hour after declaration

Answer: B
Explanation:

The stem of the question shows a dose rate increase of greater than 1000 times the normal reading for the spent fuel pool area radiation monitor. This meets the NOUE classification ion EIP-2-1 AU2 on page 46. This is also below the 2000 mR/hr condition that would require a classification of Alert for AA2 per EIP-2-1 page 51. The requirement to notify the NRC is contained in EIP-2-2, attachment 1, page 7, step 3.2.

Distracters:

A (1) This part is correct, see the explanation above. (2) The time limit given for this answer is the requirement to notify offsite authorities as stated in EIP-2-2, attachment 1, page 7, step 3.1.

C (1) This part is incorrect. The value given in the stem does not meet or exceed 2000 mR/hr which is needed to require a classification of Alert as stated in EIP-2-1 page 51. (2) Is incorrect for the same reasons as A above.

D (1) This part is incorrect for the same reason as C above. (2) This part is correct. See explanation above.

K/A Match

This question requires knowledge of the emergency plan as it pertains to emergency classification and the notification time limits.

Technical References:

EIP-2-1, CLASSIFICATION OF EMERGENCIES rev 27, EIP-2-2, CLASSIFICATION ACTIONS rev32

Handouts to be provided to the Applicants during exam:

EIP-2-1 Attachments 1, 2 and 3 (p. 13-24 of 144)

Learning Objective:

RCBT-EP-SRORMED, Objective (6): State the conditions and time limit to classify an event. (6)

Question Source:	**Bank #**	
	Modified Bank #	

	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(4)	
Level of Difficulty:	2	
SRO Only Justification: The SRO will be required to determine that the give rise in radiation is greater than 1000 times normal but less than 2000 mR/hr by selecting the correct section of EIP-2-1 and also know the notification time requirements stated in EIP-2-2		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	SRO
295024 High Drywell Pressure EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: EA2.03 Suppression pool level	**Tier**	1
	Group #	1
	K/A	295024 EA2.03
	Rating	3.8
	Revision	0
Revision Statement:		

Question: 87

Per Tech Spec 3.6.5.4, (while in mode 1) the highest limit on drywell-to-primary containment differential pressure is __(1)__ psid.

This differential pressure limit is required to ensure __(2)__.

	(1)	(2)
A.	1.2	vent clearing does not occur during normal operation
B.	1.2	suppression pool water is not forced over the weir wall
C.	1.68	vent clearing does not occur during normal operation
D.	1.68	suppression pool water is not forced over the weir wall

Answer: A

Explanation:

Per Tech Specs 3.6.5.4, Drywell Pressure, while in Modes 1, 2, and 3 Drywell-to-primary containment differential pressure shall be ≥-0.3 psid and ≤1.2 psid. Per Tech Spec Bases for 3.6.5.4, the limitation on positive drywell-to-primary containment differential pressure helps ensure that the horizontal vents are not cleared with normal weir annulus water level. The limit of -0.3 to 1.2 is for both of the listed items; however, the positive D/P limit is to prevent vent clearing during normal operation. The negative D/P limit is to prevent water over the weir wall

Distracters:

The 1.68 psid drywell-to-primary containment differential pressure is plausible because it is the EOP-1, RPV Control, entry condition.

Per Tech Spec Bases for 3.6.5.4, a negative drywell-to-primary containment differential pressure could result in overflow over the weir wall. If the applicant confuses the calculation of the pressure and subtracts drywell from containment or confuses the negative pressure for positive pressure bases, then this answer is plausible.
K/A Match The applicant must have the knowledge of the tech spec limit for high drywell pressure and understand the bases for the effect on suppression pool level.
Technical References: Tech Spec 3.6.5.4, Drywell Pressure, and bases EOP-1, RPV Control
Handouts to be provided to the Applicants during exam: NONE
Learning Objective: RLP-STM-57-LO Rev 0, Objective 9: Identify the technical specifications and/or technical requirements manual requirements related to primary containment. (9)

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	43(b)(2)	
Level of Difficulty:	3	

SRO Only Justification: The applicant must have the required knowledge of Tech Spec 3.6.5.4, Drywell Pressure, and its bases.
PRA Applicability:

Examination Outline Cross Reference	**Level**	SRO
295025 Reactor pressure G2.2.44 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.	**Tier**	1
	Group #	1
	K/A	295025 G2.2.44
	Rating	4.4
	Revision	2
Revision Statement: Rev 2: Changed question source to bank question.		

Question: 88

The Reactor has been manually scrammed.

8 minutes after the event, SRV B21-F051C is taken to OPEN. With no other operator action for pressure control, reactor pressure is observed cycling between 1063 psig and 956 psig.

Suppression pool temperature is 111°F.

Which of the following is correct?

A. Implement EOP-1, RPV Control, and stabilize RPV pressure below 1090 psig with SRVs.

B. Implement EOP-1A, RPV Control ATWS, and stabilize RPV pressure between 960 psig and 1090 psig with SRVs.

C. Implement EOP-1, RPV Control and use RWCU in the blowdown mode per Enclosure 29, RWCU Blowdown Mode, to control reactor pressure.

D. Implement EOP-1A, RPV Control ATWS and use RWCU in the blowdown mode per Enclosure 29, RWCU Blowdown Mode, to control reactor pressure.

Answer: B
Explanation: Each SRV has a rating of 6-7% steam flow. With 5 having Low Low Set mode set points of 1143-956 psig. With the given conditions, more than one SRV required to control RPV pressure, the plant is in an ATWS condition and EOP-1A, RPV Control ATWS, should be entered. SRVs should be manually controlled to maintain pressure below 1090 psig per step RPA-3.

Distracters:

A The direction given in this distractor is correct for pressure control guidance of EOP-1 step RP-2. The stem conditions indicate that the reactor is not shut down and EOP-1A should be used. An unprepared applicant may not recognize that indications show reactor power above 5% which is an entry condition for EOP-1A

C EOP-1 is an incorrect answer for reasons given in A above. Use of RWCU in the blow down mode for pressure control is plausible since it is a pressure control method given in Table P-1 of EOP-1 and referenced by step RP-2 of EOP-1.

D EOP-1A is correct for reasons given in B above. Use of RWCU in the blow down mode for pressure control is plausible since it is a pressure control method given in Table P-1A of EOP-1A however Table P-1A systems are not allowed for use(step RPA-5) until pressure has been stabilized per step RPA-3. Per EOP-1A, Enclosure 29 is only allowed if boron has not injected. Suppression pool temperature is greater than 110°F, per EOP-1A, step RQA-4 boron should be injected.

K/A Match

The control room indications given in the question will test the applicant ability to direct system operation that affects plant systems and high pressure condition.

Technical References:

EOP-1A, Rev 28, EOP-1, Rev 28, RSTM..0109 MAIN STEAM SYSTEM, Rev 015 pages 9, 12, 61

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-HLO-512, Objective 7: Given EOP-1 RPV Control and EOP-4 Contingencies flowcharts and plant conditions, determine the next action required to be implemented.

Question Source:	Bank # 2008 NRC #79	X
	Modified Bank #	
	New	

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: The applicant must assess the conditions in the stem to determine that the plant is not shut down and EOP-1A entry is required		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA2. Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EA2.02 Reactor water level	**Tier**	1
	Group #	1
	K/A	295037 EA2.02
	Rating	4.2
	Revision	1
Revision Statement: Rev 1: Changed stem to state reactor scram has occurred, instead of EOP-1A was entered.		

Question: 89

Reactor scram has occurred.

Reactor power is 9%.

RPV water level is +25 inches Narrow Range.

Reactor pressure is 950 psig with the Pressure Regulator controlling Reactor Pressure.

What RPV water level band should be maintained?

A. -20 to 51 inches

B. 10 to 51 inches

C. -60 to -140 inches

D. -100 to -140 inches

Answer: C
Explanation: EOP-1A, RPV Control – ATWS is entered due to reactor scram and reactor power is 9%. The Level/Power leg is started. Pressure regulator is controlling pressure, so the MSLs are open. Reactor power is 9%. RPV water level is +25 inches. Currently there

are no challenges to the Suppression Pool or drywell. Per OSP-53, Attachment 1, Level Control Strategy Summary Page, during an ATWS with reactor power greater than 5% with no energy rejection to the suppression pool, the RPV level control band is -60 to -140 inches.

Distracters:

A. Per OSP-53, Attachment 1A, Post Scram Level control Strategies, the non-ATSW (expanded) band is -20 to 51 inches.
B. Per OSP-53, Attachment 1A, Post Scram Level control Strategies, the ATWS with reactor power less than 5% level band is 10 to 51 inches.
D. Per OSP-53, Attachment 1A, Post Scram Level control Strategies, during an ATWS if reactor power is greater than 5% and Level/Power control conditions are anticipated, the RPV level control band is -100 to -140 inches. There is no indication of heat addition to the suppression pool. The MSIVs remain open; therefore, no heat is being added to the suppression pool. This is plausible if applicant believes heat is being added to the suppression pool.

K/A Match

The applicant must diagnose from the given plant information that the ATWS is >5% and requires a reduction in RPV water level per EOP-1A, RPV Control – ATWS.

Technical References:

EOP-1A, RPV Control – ATWS, Revision 28
OSP-53, Emergency and Transient Response Support Procedure, Revision 25

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-HLO-512 Objective 8: Given EOP-1 RPV Control and EOP-4 Contingencies flowcharts and plant conditions, accurately apply the strategies provided in OSP-0053, Emergency and Transient Response Support Procedure

Question Source:	**Bank #** RBS-OPS-06327	X
	Modified Bank #	
	New	

Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(5)	
Level of Difficulty:	3	

SRO Only Justification:

Applicant must assess the given plant conditions, then select the appropriate RPV level band to mitigate the ATWS.

PRA Applicability:

Examination Outline Cross Reference	Level	SRO
700000 Generator Voltage and Electric Grid Disturbances AA2. Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AA2.07 Operational status of engineered safety features	**Tier**	1
	Group #	1
	K/A	700000 AA2.07
	Rating	4.0
	Revision	0
Revision Statement:		

Question: 90

A grid disturbance has occurred.

Current plant conditions are:

- 100% rated power
- 50 MVARSOUT
- Alarm P808-86A-H01, GRID TROUBLE is received in the MCR
- Bus voltage on SPI-REC102 indicates 223 KV
- The SOC has notified the OSM/CRS that post-trip switchyard voltage does not support accident loading

___(1)___ should be entered and offsite power system should be declared ___(2)___.

	(1)	(2)
A	AOP-4, LOSS OF OFFSITE POWER	INOPERABLE
B	AOP-64, DEGRADED GRID	EQUPMENT NON-FUNCTIONAL
C	AOP-4, LOSS OF OFFSITE POWER	EQUPMENT NON-FUNCTIONAL
D	AOP-64, DEGRADED GRID	INOPERABLE

Answer: D
Explanation:

Correct.AOP-64 is the correct procedure to be entered. The purpose of this AOP is to provide instructions in the event of a degraded grid condition of the Entergy transmission grid. AOP-64 p3. Page 12 directs the SRO to declare the offsite power system inoperable under the conditions stated in the stem of the question.

Distracters:

(1) AOP-004 gives guidance during a loss of offsite power source. While the stem notes only a degraded condition of the offsite power source, not a loss. The directions for this distractor are correct for AOP-4(p16) but not given in AOP-64. An unprepared student may choose this answer.

(2) Per EN-OP-104, EQUIPMENT NON-FUNCTIONAL is a condition where a Non-TS SSC is not functional. Plausible if applicant confuses grid disturbance with non-TS SSC.

K/A Match

The applicant must determine operability status of ESF features due to switchyard low voltage during a grid disturbance.

Technical References:

AOP-0064 Degraded Grid rev 10,
AOP-4, loss of offsite power rev 61.
TS 3.8.1
EN-OP-104,

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-OPS-AOP0064, Objective 2: Given event/transient symptoms, determine if AOP-0064 implementation is required. (2)

Question Source:	**Bank #**	
	Modified Bank #	

	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(2) & (5)	
Level of Difficulty:	3	
SRO Only Justification: The applicant must assess the conditions in the stem to determine that AOP-64 is the proper procedure to enter and declare the offsite power system INOPERABLE as required by TS3.8.1 D.		
PRA Applicability:		
Top 10 Risk Significant Systems: 230 kV AC Power		

Examination Outline Cross Reference	Level	SRO
295009 Low Reactor Water Level 2.4.6 Knowledge of EOP mitigation strategies.	Tier	1
	Group #	2
	K/A	295009 G.2.4.6
	Rating	4.7
	Revision	1
Revision Statement: Rev 1: Changed the word “might” to “will.”		

Question: 91

Which of the following strategies will be employed while carrying out the steps of EOP-1A, RPV Control – ATWS?

A. After SRVs are opened for Emergency Depressurization, inject with all available injection sources to restore and maintain RPV level above -162 inches minimum.

B. After SRVs are opened for Emergency Depressurization AND when RPV pressure has lowered below MSCP, commence and slowly raise injection into the RPV to restore and maintain RPV level above -187 inches minimum.

C. With no injection source available, steam cooling is required; delay emergency depressurization until RPV level drops to -200 inches.

D. After injection of cold shutdown boron weight has been verified by a Reactor Operator, Exit EOP-1A, RPV Control ATWS and enter EOP-1, RPV Control.

Answer: B

Explanation:

Per EOP-1A, RPV Control – ATWS, Steps RLA-20 and RLA21, When RPV pressure is below the Minimum Steam Cooling Pressure, commence and slowly raise injection into the RPV to restore and maintain RPV level above -187 inches. RLA20 is a stop sign and conditions must be properly diagnosed by the SRO to direct further actions of the EOP.

Distracters:

A. Per EOP-1, RPV Control, Step ALC-12, after Emergency Depressurization, restore and maintain RPV level above -162 inches with any Table L-1 or L-2 systems. This plausible if applicant confuses ATWS strategy with non-ATWS strategy.

C. Per EOP-1, Step ALC-7, if NO injection source is lined up with a pump running then steam cooling is required. Per steps STC-3 and 4, when RPV level drops to -200 inches, Emergency Depressurization is required. This plausible if applicant confuses ATWS strategy with non-ATWS strategy.

D. Plausible if applicant does not recall that EOP-1A cannot be exited until the reactor will remain shutdown under all conditions without boron (ie. Control rods)

K/A Match

Applicant must have knowledge of EOP-1A, RPV Control – ATWS, RPV level control and Emergency Depressurization requirement and method.

Technical References:

EOP-1 and EOP 1A Revision 28.

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-LO-513, Objective 5: Given flowcharts for EOP-1A, RPV Control – ATWS, and EOP-4A, Contingencies – ATWS, and plant conditions; determine the next action required to be implemented.

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	

SRO Only Justification:

Applicant must assess plant conditions during ATWS and select the section of the procedure to mitigate the low RPV water level. The applicant must also have knowledge of the EOP decision point to Emergency Depressurize during an ATWS and

what plant conditions must be met to restore RPV level.
PRA Applicability:
Top 10 Operator Actions: Manual depressurization of reactor vessel

Examination Outline Cross Reference	**Level**	SRO
295014 Inadvertent Reactivity Addition AA2 - Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION AA2.01 Reactor power.	**Tier**	1
	Group #	2
	K/A	295014 AA.01
	Rating	4.2
	Revision	3
Revision Statement: Capitalized The word “steam” in distractor B. Replaced distractor B so that there would not be 3 distractors that contain “main steam” Although C & D both contain “main steam line”, one is associated with pressure and other is associated with flow which are mutually exclusive.		

Question: 92

Per Tech Spec bases, which RPS Instrumentation function is assumed for over pressurization protection analysis to limit the peak reactor pressure vessel pressure to less than the ASME Code limits?

A. Average Power Range Monitor Fixed Neutron Flux

B. Reactor Vessel Water Level

C. Main Steam Line Pressure

D. Main Steam Line Flow

Answer: A
Explanation: Per Tech Spec Bases B3.3.1.1 RPS Instrumentation, the Average Power Range Monitor Fixed Neutron Flux-High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 2, the Average Power Range Monitor Fixed Neutron Flux-High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could

challenge the integrity of the fuel cladding and the RCPB. No specific safety analysis takes direct credit for this Function. However, the Reactor Vessel Steam Dome Pressure-High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 2, the reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux-High signal, not the Reactor Vessel Steam Dome Pressure-High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

Distracters:

B. Plausible if applicant mistakenly concludes that on rising reactor water level the combined reactor power increase from "a significant amount of relatively cold feedwater" (B 3.3.1.1-5) along with the reduced steam dome volume from the rising level would result in a limiting overpressure transient of the reactor vessel.

C. Plausible if applicant confuses Primary Containment and Drywell Isolation Instrumentation B3.3.6.1 1b.Main Steam Line Pressure-Low, The Main Steam Line Pressure-Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 685 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 23.8% RTP.)

D. Plausible if applicant confuses Primary Containment and Drywell Isolation Instrumentation B3.3.6.1 1.c. Main Steam Line Flow-High. The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 50.67 limits.

K/A Match

The applicant must have the knowledge that the inadvertent addition of reactivity due to pressure transient is limited by the Average Power Range Monitor Fixed Neutron Flux-High. The applicant must have the understanding of how the pressure transient inserts reactivity and raises reactor power.

Technical References:

Tech Spec Bases 3.3.1.1 RPS Instrumentation
Tech Spec Bases 3.3.6.1, Primary Containment and Drywell Isolation Instrumentation

Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-0508, Objective (10): Identify the Technical Specifications and/or Technical Requirements Manual requirements of the Reactor Protection System. (10)		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification: The applicant must have knowledge of TS bases for RPS instrumentation.		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
295017 High Off-Site Release Rate AA2. Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : AA2.01 †Off-site release rate: Plant-Specific	Tier	1
	Group #	2
	K/A	295017 AA2.01
	Rating	4.2
	Revision	0
Revision Statement:		

Question: 93

Operators have entered EOP-1 (RPV Control) and EOP-3 (Secondary Containment and Radioactivity Release Control) due to a primary system discharging outside primary containment.

The discharge cannot be isolated.

Per EOP-3, before the offsite release rate reaches a(n) ___(1)___, the crew is required to ___(2)___.

	(1)	(2)
A.	SAE	reduce RPV pressure band to 500 to 700 psig
B.	SAE	emergency depressurize
C.	General Emergency	reduce RPV pressure band to 500 to 700 psig
D.	General Emergency	emergency depressurize

Answer: D
Explanation: Per EOP-3, Secondary Containment and Radioactivity Release Control, steps RR-4 and RR-5, before the offsite radioactivity release rate reaches a General Emergency (EIP-2-001), Emergency Depressurization is required.

Distracters:
SAE is plausible because it is one level below General Emergency when declaring Emergencies. Applicant may confuse General Emergency requirement with SAE.

Reducing pressure is plausible because OSP-53, Attachment 1B, Post Scram Pressure Control Strategies, states In the condition where there is an unisolable reactor coolant leak, it is appropriate to lower the RPV pressure to reduce the driving head of the leak. The CRS then changes the pressure band to 500 to 700 psig only when pressure is stable and controllable.

K/A Match

The applicant must demonstrate knowledge of EOP-3 decision point for offsite release rate. The applicant must recall the level at which action is taken and direct the correct action to mitigate the release.

Technical References:

EOP-3, Revision 18
OSP-53, Revision 25

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-LO-0515 Rev 0, Objective 6: Given EOP-3 flowchart and plant conditions, determine the next action required to be implemented.

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	

SRO Only Justification: The applicant must have the knowledge of EOP-3 diagnostic step and decision point to transition to emergency depressurization based on high offsite release rate.
PRA Applicability:

Examination Outline Cross Reference	Level	SRO
2.1.1 Knowledge of conduct of operations requirements.	Tier	3
	Group #	N/A
	K/A	2.1.1
	Rating	4.2
	Revision	0
Revision Statement:		

Question: 94

Per EN-OP-115 (Conduct of Operations), it is a responsibility of the Shift Manager to “remain within ______ minutes of the Control Room and immediately return to the Control Room to provide oversight of activities during accident or abnormal conditions.”

A. 5

B. 10

C. 15

D. 30

Answer: B
Explanation: Per EN-OP-115, Conduct of Operations, Remain within ten minutes of the Control Room and immediately return to the Control Room and provide oversight of activities during accident or abnormal conditions.
Distracters: 5, 15, and 30 minutes are plausible if applicant confuses time requirement for the Shift Manager to be able to return to the Control Room.
K/A Match Applicant must have knowledge of EN-OP-115, Conduct of Operations, time requirements for shift manager to report to the Control Room.

Technical References: EN-OP-115, Conduct of Operations, Revision 23		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-HLO-206, Objective 2: List ten responsibilities of the OSM (2).		
Question Source:	**Bank # GGNS 2011 NRC**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: Applicant must have knowledge requirements of administrative procedures including roles and responsibilities of the shift manager.		
PRA Applicability:		

Examination Outline Cross Reference	**Level**	SRO
2.1.40 Knowledge of refueling administrative requirements.	**Tier**	3
	Group #	N/A
	K/A	2.1.40
	Rating	3.9
	Revision	5
Revision Statement: Revised stem to combine two sentences into one.		

Question: 95

Per FHP-1, CONTROL OF FUEL HANDLING AND REFUELING OPERATIONS:

The ______ has the authority to omit, bypass, or combine steps as necessary providing it is documented, unless it is specifically stated in the procedure for those steps to be followed step by step as written(for actions that do not affect safety systems, limits or functions).

A. Refuel SRO

B. Refuel Floor Supervisor

C. Operations Shift Manager

D. Operations Manager

Answer: C

Explanation:

Per FHP-1, CONTROL OF FUEL HANDLING AND REFUELING OPERATIONS, 2.2 Depending on actual plant conditions, the Operations Shift Manager has the authority to omit, bypass, or combine steps as necessary providing it is documented, unless it is specifically stated in the procedure for those steps to be followed step by step as written.

Distracters:

All distractors are plausible based on roles and responsibilities during refuel operations.

K/A Match Applicant must have knowledge of who has authority to omit, bypass, or combine steps in the fuel handling procedure.		
Technical References: FHP-1, CONTROL OF FUEL HANDLING AND REFUELING OPERATIONS, Revision 36		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-STM-55, Refueling Systems, Objective (6): Given a Precaution & Limitation from FHP-0001 Control of Fuel Handling and Refueling, FHP-0002 Fuel Handling Platform, FHP-0003 Refueling Platform, FHP-0005 Fuel Transfer Tube, or FHP-0008 Fuel Transfer Tube Operation in Modes 1, 2, or 3, explain its basis.		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	43(b)(7)	
Level of Difficulty:	3	
SRO Only Justification: Applicant must have required knowledge of precaution and limitation stated in fuel handling procedure.		
PRA Applicability:		

Examination Outline Cross Reference	Level	SRO
2.2.12 Knowledge of surveillance procedures.	Tier	3
	Group #	N/A
	K/A	2.2.12
	Rating	4.1
	Revision	1
Revision Statement: Revised stem to read, “what is the latest time the surveillance may be completed to comply with Tech Specs?”		

Question: 96

An STP has a DUE DATE of 11/1/2012 @ 2230.

The actual performance was completed and signed off on 11/1/2012 at 2215.

If the surveillance frequency is 24 hours, what is the latest time the surveillance may be completed to comply with Tech Specs?

(1) DUE DATE (2) LATE DATE

A. (1) 11/2/2012 @ 2215; (2) 11/3/2012 @ 0415

B. (1) 11/2/2012 @ 2230; (2) 11/3/2012 @ 0430

C. (1) 11/2/2012 @ 2230; (2) 11/3/2012 @ 0230

D. (1) 11/2/2012 @ 2215; (2) 11/2/2012 @ 2230

Answer: A

Explanation:

Due date is 24 hours from completion of last performance. Late date includes 1.25 tolerance per Tech Spec S.R.3.0.2.

Distracters:

B. Next performance is based on completion time of previous performance; not on previous due dates and late dates.

C. In this case the Due Date is 24 hours and 15 minutes from the last performance. Due Date is 24 hours from the previous performance. The Late Date would be 1.25 times the normal frequency or 30 hours from the previous performance. The Late Date in part 2 is 28.25 hours from the previous performance. D. Part 1 is correct, but the Part 2 answer did not taken into account the 1.25 tolerance allowance.		
K/A Match Applicant must have knowledge of the Tech Spec time requirements for completing surveillances.		
Technical References: Tech Spec SR 3.0.2 ADM-0015 Rev 36 Pg 4-5 of 42		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RLP-OPS-HO221 Objective 5: Given a Surveillance Test Look Ahead Report, determine the due date and late date.		
Question Source:	**Bank # 2012 NRC Q96**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification:		

Applicant must apply Tech Spec rules per section 4 to determine proper completion and late times for the given surveillance.
PRA Applicability:

Examination Outline Cross Reference	**Level**	SRO
2.2.37 Ability to determine operability and/or availability of safety related equipment.	**Tier**	3
	Group #	N/A
	K/A	G 2.2.37
	Rating	4.6
	Revision	1

Revision Statement:
Replaced “Equipment Functional” with “OPERABLE-DNC” and updated distractor explanation.

Question: 97

A Condition Report (CR) has been initiated to document housekeeping deficiencies in a room containing Engineered Safety Feature (ESF) system components.

A scaffold containing a large, unsecured toolbox was found next to the ESF system injection valve and the toolbox has not yet been removed.

What Operability Code should be assigned to the initial screening of this CR in PCRS?

A. INOPERABLE

B. NOT REQUIRED

C. OPERABLE-DNC

D. EQUIPMENT NON-FUNCTIONAL

Answer: A

Explanation:

EN-OP-104, Operability Determination Process, INOPERABLE is a condition where a TS SSC is not OPERABLE. All ESF equipment is covered by Tech Specs. Attachment 9.1, Operability Classification Guide, states that analysis or evaluation has determined that a condition, in conjunction with a credible Consequential Failure during a DBE would result in the loss of a Safety Function. If a DBE earthquake occurred the injection valve would be damaged. Therefore the classification is required to be INOPERABLE.

Distracters:

B. Per EN-OP-104, this is an Operability Code used in PCRS for SSCs that are not in the scope for either Operability determination or Functionality assessment. This is plausible because the applicant may believe the scaffolding and toolbox is not in the scope.
C. Per EN-OP-104, this is a condition where a TS SSC is OPERABLE but a Degraded or Nonconforming Condition exists that does not require Compensatory Measures.
D. Per EN-OP-104, this is a condition where a Non-TS SSC is not functional. This is plausible if applicant does not recognize the TS applicability.

K/A Match

Applicant must have knowledge of the effect the scaffolding and unsecured toolbox will have on operability of the ESF system. Knowledge is also required to know that equipment covered by tech specs is OPERABLE or INOPERABLE.

Technical References:

EN-OP-104, Operability Determination Process, Revision 13

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-HLO-416, Objective 3: 3. Give a working definition of the following terms in accordance with Technical Specifications:
k) Operable - Operability

Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	
	Comprehensive / Analysis	X
10CFR Part 55 Content:	43(b)(2)	
Level of Difficulty:	3	
SRO Only Justification:		

The applicant must have knowledge of Tech Spec bases that is required to maintain ESF systems operable. Also, the applicant must have knowledge of Operability determination process which is only performed by an SRO.
PRA Applicability:

Examination Outline Cross Reference	Level	SRO
2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	Tier	3
	Group #	N/A
	K/A	2.3.13
	Rating	3.8
	Revision	0
Revision Statement:		

Question: 98

River Bend Station is currently performing a refueling outage with core reload in progress.

A control rod blade guide must be moved from the core to the wall hangers in the upper pool. Due to the length of the blade guide, the mast must be raised beyond the HOIST UP position while traversing through the Portable Shielding (Cattle Chute).

Per EN-FAP-OU-108, Fuel Handling Process, who must provide approval authority to allow the Refuel Bridge Driver to utilize the TRAVEL OVERRIDE and HOIST OVERRIDE interlock bypass features to move control rod blade guides through the Cattle Chute?

A. Control Room Supervisor

B. Refuel Floor Supervisor

C. Refuel SRO

D. Fuel Movement Supervisor

Answer: C

Explanation:

Per EN-FAP-OU-108, Fuel Handling Process, Roles and Responsibilities, the Fuel Handling Supervisor provides approval authority to Fuel Handling Crew personnel to bypass equipment interlocks as authorized by site specific procedures (Licensed SRO function only).

Distracters:

All other supervisors are plausible because they are involved in the fuel handling.

K/A Match

The applicant must have knowledge of the fuel handling procedures and roles and responsibilities for overriding interlocks that ensure radiological safety during refuel operations.

Technical References:

EN-FAP-OU-108, Fuel Handling Process

Handouts to be provided to the Applicants during exam:

NONE

Learning Objective:

RLP-STM-0055 Obj. 6: Given a Precaution & Limitation from FHP-0001 Control of Fuel Handling and Refueling, FHP-0002 Fuel Handling Platform, FHP-0003 Refueling Platform, FHP-0005 Fuel Transfer Tube, or FHP-0008 Fuel Transfer Tube Operation in Modes 1, 2, or 3, explain its basis. (6)

Question Source:	**Bank # 2010 Audit**	X
	Modified Bank #	
	New	
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(7)	
Level of Difficulty:	3	

SRO Only Justification:

The applicant must have knowledge of the refuel SRO responsibilities and authority to override interlocks.

PRA Applicability:

Examination Outline Cross Reference	Level	SRO
2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.	Tier	3
	Group #	N/A
	K/A	G.2.4.22
	Rating	4.4
	Revision	1

Revision Statement:
Replaced C with “avoid an unnecessary transient, down power, or shutdown” which describes circumstances that could justify application for a Notice of Enforcement Discretion (NOED).

Question: 99

Per 10CFR50.54(x), 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. avoid an unnecessary transient, down power, or shutdown

D. extend safety related equipment out of service time.

Answer: B

Explanation:

Per 10CFR50.54 Conditions of licenses, part (x) A licensee may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under this part) in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent.

Distracters:

A. Plausible if applicant confuses 10CFR50.54(x) with the ability to extend operational exposure limits in 10CFR part 20.
C. Plausible if applicant confuses 10CFR50.54(x) with a Notice of Enforcement Discretion (NOED). NRC IM 0410 NOTICES OF ENFORCEMENT DISCRETION

D. Plausible if required to adjust maintenance rule outage time.

K/A Match

The applicant must have knowledge of the requirements allowed to prioritize safety function actions to protect the health and safety of the public. Knowledge to allow deviation from licensing procedures to prioritize safety functions is required.

Technical References:

10CFR50.54x, EN-OP-115 Rev 14 Pg 5 of 89
NRC IM 0410 NOTICES OF ENFORCEMENT DISCRETION

Handouts to be provided to the Applicants during exam:
NONE

Learning Objective:

RLP-HLO-0206 Obj 16: State the actions that must be taken when a procedure cannot or should not be followed as written.

Question Source:	**Bank # March 2014 NRC Q99**	X
	Modified Bank #	
	New	

Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	

10CFR Part 55 Content:	55.43(b)(1)	

Level of Difficulty:	2	

SRO Only Justification:
The applicant must have knowledge of the administrative procedure to allow deviations from a license condition, per 10CFR50.54x.

PRA Applicability:

Examination Outline Cross Reference	Level	SRO
2.4.40 Knowledge of SRO responsibilities in emergency plan implementation.	Tier	3
	Group #	N/A
	K/A	2.4.40
	Rating	4.5
	Revision	1
Revision Statement: Replaced 15 minutes with 1 hour in A & C.		

Question: 100

The Long Notification Message Form (LNMF) is sent out for any significant changes to ___(1)___.

No more than ___(2)___ should be exceeded between any two LNMFs.
PAR – Protective Action Recommendations

	(1)	(2)
A.	PAR	1 hour
B.	PAR	2 hours
C.	plant conditions	1 hour
D.	plant conditions	2 hours

Answer: D

Explanation:

Per EIP-2-006, Emergency Implementing Procedure, the Long Notification Message Form (LNMF) is used for providing State and local authorities follow-up information. The LNMF is sent out as soon as possible following a SNMF. The LNMF is also sent out for any significant changes to plant conditions that do not require an emergency escalation or change in PARs. No more than 2 hours should be exceeded between any two LNMFs.

Distracters:

Per EIP-2-002, the NRC is required to be notified immediately after notifiying state and local authorities and not later than one hour after declaring the emergency. Distractors are plausible if applicant confuses requirements for the SNMF and LNMF.		
K/A Match Applicant must have knowledge of SM responsibilities and time requirements associated with the LNMF during a plant emergency.		
Technical References: Per EIP-2-006, Rev 44		
Handouts to be provided to the Applicants during exam: NONE		
Learning Objective: RCBT-EP-SRORMED, Objective 6: State the conditions and time limit to classify an event. (6)		
Question Source:	**Bank #**	
	Modified Bank #	
	New	X
Question Cognitive Level:	**Memory / Fundamental**	X
	Comprehensive / Analysis	
10CFR Part 55 Content:	55.43(b)(5)	
Level of Difficulty:	3	
SRO Only Justification: Applicant must have knowledge of emergency plan criteria and regulator limits during a plant emergency that requires a LNMF and update.		

PRA Applicability:

MODE SELECTION

1 Time Since Reactor Shutdown

Record current date and time ____________/____________

Record date and time of Reactor shutdown _____________

Determine length of time since Reactor shutdown ______________ Hours

2 Reactor Core Decay Heat from Attachment 6 or Incore Fuels Group

____________________10^6 BTU/HR

Attachment 6 / Incore Fuels Group (Circle one)

MODE SELECTION

NOTE

An Alternate Method is required for each inoperable RHR-SDC mode.

More than one system may be needed to meet Reactor Core Decay Heat requirements; however, credit for each system may be used only once.

3 Compare Reactor Core Decay Heat value from Step **2** of this attachment to the systems heat removal capacities below and determine the Alternate Shutdown Cooling Methods. Record Alternate Shutdown Cooling Methods in Step **6.2.3.2**.

SYSTEM	HEAT REMOVAL CAPACITY (BTU/HR)	CONDITIONS / ASSUMPTIONS
SPC/ADHR	37.67×10^6 40.12×10^6 70.86×10^6	2500 gpm, 83°F Service Water & 2250 gpm, 120°F Rx Coolant 2250 gpm, 140°F Rx Coolant 2250 gpm, 200°F Rx Coolant
FPC Assist	See Attachment 2	2000 gpm, 95°F Service Water
CRD	2.5×10^6	50 gpm, 100°F CRD & 200°F Rx Coolant
Condensate	7.5×10^6	200 gpm, 124°F Condensate & 200°F Rx Coolant
RWCU	See Attachment 3	471 gpm RPCCW** & 248 gpm Rx Coolant
RHR-LPCI	126×10^6	5800 gpm, 95°F Service Water & 5050 gpm, 185°F Rx Coolant
SFC	See Attachment 4	2000 gpm Service Water & 2500 gpm Rx Coolant
MSL Flooding	60×10^6	1758 gpm/20 hrs after shutdown &, 120°F to 170°F Rx Coolant &, 65°F to 100°F Circulating Water

** This pertains to the non-regenerative heat exchanger only.

FPC ASSIST TEMPERATURE DEPENDENT HEAT REMOVAL CAPACITY TABLE

FPC Assist of RHR, Service Water Flow @ 2000GPM							
SW	Heat Removal Capacity (MBtu/hr)						
Tci (°F)	Pool = 95°F	Pool = 100°F	Pool = 105°F	Pool = 110°F	Pool = 120°F	Pool = 130°F	Pool = 140°F
75	11.5	19.17	23.00	26.84	34.50	42.17	49.84
77	9.97	17.64	21.47	25.30	32.97	40.64	48.31
79	8.43	16.10	19.94	23.77	31.44	39.11	46.77
81	6.90	14.57	18.40	22.24	29.90	37.57	45.24
83	5.37	13.04	16.87	20.70	28.37	36.04	43.71
85	3.83	11.50	15.34	19.17	26.84	34.50	42.17
87	2.30	9.97	13.80	17.64	25.30	32.97	40.64
89	.77	8.43	12.27	16.10	23.77	31.44	39.11
91		6.90	10.73	14.57	22.24	29.90	37.57
93		5.37	9.20	13.04	20.70	28.37	36.04
95		3.83	7.67	11.50	19.17	26.84	34.50
97		2.30	6.13	9.97	17.64	25.30	32.97
99		.77	4.60	8.43	16.10	23.77	31.44
101			3.07	6.90	14.57	22.24	29.90
103			1.53	5.37	13.04	20.70	28.37
105				3.83	11.5	19.17	26.84

FPC ASSIST OF RHR, SERVICE WATER FLOW @ 2000GPM						
SW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Pool = 150°F	Pool = 160°F	Pool = 170°F	Pool = 180°F	Pool = 190°F	Pool = 200°F
75	57.51	65.18	72.84	80.51	88.18	95.85
77	55.97	63.64	71.31	78.98	86.65	94.31
79	54.44	62.11	69.78	77.44	85.11	92.78
81	52.91	60.57	68.24	75.91	83.58	91.25
83	51.37	59.04	66.71	74.38	82.04	89.71
85	49.84	57.51	65.18	72.84	80.51	88.18
87	48.31	55.97	63.64	71.31	78.98	86.65
89	46.77	54.44	62.11	69.78	77.44	85.11
91	45.24	52.91	60.57	68.24	75.91	83.58
93	43.71	51.37	59.04	66.71	74.38	82.04
95	42.17	49.84	57.51	65.18	72.84	80.51
97	40.64	48.31	55.97	63.64	71.31	78.98
99	39.11	46.77	54.44	62.11	69.78	77.44
101	37.57	45.24	52.91	60.57	68.24	75.91
103	36.04	43.71	51.37	59.04	66.71	74.38
105	34.50	42.17	49.84	57.51	65.18	72.84

RWCU TEMPERATURE DEPENDENT HEAT REMOVAL CAPACITY TABLES

RWCU						
RPCCW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 90°F	Rx = 100°F	Rx = 110°F	Rx = 120°F	Rx = 130°F	Rx = 140°F
75	1.48	2.47	3.46	4.44	5.43	6.42
77	1.28	2.27	3.26	4.25	5.23	6.22
79	1.09	2.07	3.06	4.05	5.04	6.03
81	0.89	1.88	2.86	3.85	4.84	5.83
83	0.69	1.68	2.67	3.65	4.64	5.63
85	0.49	1.48	2.47	3.46	4.44	5.43
87	0.30	1.28	2.27	3.26	4.25	5.23
89	0.10	1.09	2.07	3.06	4.05	5.04
91		0.89	1.88	2.86	3.85	4.84
93		0.69	1.68	2.67	3.65	4.64
95		0.49	1.48	2.47	3.46	4.44
97		0.30	1.28	2.27	3.26	4.25
99		0.10	1.09	2.07	3.06	4.05
101			0.89	1.88	2.86	3.85
103			0.69	1.68	2.67	3.65
105			0.49	1.48	2.47	3.46

RWCU						
RPCCW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 150°F	Rx = 160°F	Rx = 170°F	Rx = 180°F	Rx = 190°F	Rx = 200°F
75	7.41	8.40	9.38	10.37	11.36	12.35
77	7.21	8.20	9.19	10.17	11.16	12.15
79	7.01	8.00	8.99	9.98	10.96	11.95
81	6.82	7.80	8.79	9.78	10.77	11.75
83	6.62	7.61	8.59	9.58	10.57	11.56
85	6.42	7.41	8.40	9.38	10.37	11.36
87	6.22	7.21	8.20	9.19	10.17	11.16
89	6.03	7.01	8.00	8.99	9.98	10.96
91	5.83	6.82	7.80	8.79	9.78	10.77
93	5.63	6.62	7.61	8.59	9.58	10.57
95	5.43	6.42	7.41	8.40	9.38	10.37
97	5.23	6.22	7.21	8.20	9.19	10.17
99	5.04	6.03	7.01	8.00	8.99	9.98
101	4.84	5.83	6.82	7.80	8.79	9.78
103	4.64	5.63	6.62	7.61	8.59	9.58
105	4.44	5.43	6.42	7.41	8.40	9.38

SFC TEMPERATURE DEPENDENT HEAT REMOVAL CAPACITY TABLES

SSW or RPCCW	SFC Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 90°F	Rx = 100°F	Rx = 110°F	Rx = 120°F	Rx = 130°F	Rx = 140°F
75	6.96	11.61	16.25	20.89	25.54	30.18
77	6.04	10.68	15.32	19.97	24.61	29.25
79	5.11	9.75	14.39	19.04	23.68	28.32
81	4.18	8.82	13.47	18.11	22.75	27.39
83	3.25	7.89	12.54	17.18	21.82	26.47
85	2.32	6.96	11.61	16.25	20.89	25.54
87	1.39	6.04	10.68	15.32	19.97	24.61
89	0.46	5.11	9.75	14.39	19.04	23.68
91		4.18	8.82	13.47	18.11	22.75
93		3.25	7.89	12.54	17.18	21.82
95		2.32	6.96	11.61	16.25	20.89
97		1.39	6.04	10.68	15.32	19.97
99		0.46	5.11	9.75	14.39	19.04
101			4.18	8.82	13.47	18.11
103			3.25	7.89	12.54	17.18
105			2.32	6.96	11.61	16.25

SSW or RPCCW	SFC Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 150°F	Rx = 160°F	Rx = 170°F	Rx = 180°F	Rx = 190°F	Rx = 200°F
75	34.82	39.47	44.11	48.75	53.40	58.04
77	33.90	38.54	43.18	47.82	52.47	57.11
79	32.97	37.61	42.25	46.90	51.54	56.18
81	32.04	36.68	41.32	45.97	50.61	55.25
83	31.11	35.75	40.40	45.04	49.68	54.33
85	30.18	34.82	39.47	44.11	48.75	53.40
87	29.25	33.90	38.54	43.18	47.82	52.47
89	28.32	32.97	37.61	42.25	46.90	51.54
91	27.39	32.04	36.68	41.32	45.97	50.61
93	26.47	31.11	35.75	40.40	45.04	49.68
95	25.54	30.18	34.82	39.47	44.11	48.75
97	24.61	29.25	33.90	38.54	43.18	47.82
99	23.68	28.32	32.97	37.61	42.25	46.90
101	22.75	27.39	32.04	36.68	41.32	45.97
103	21.82	26.47	31.11	35.75	40.40	45.04
105	20.89	25.54	30.18	34.82	39.47	44.11

COMBINED RWCU AND SFC TEMPERATURE DEPENDENT HEAT REMOVAL CAPACITY TABLES

RWCU and SFC						
RPCCW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 90°F	Rx = 100°F	Rx = 110°F	Rx = 120°F	Rx = 130°F	Rx = 140°F
75	8.45	14.08	19.71	25.34	30.97	36.60
77	7.32	12.95	18.58	24.21	29.84	35.47
79	6.19	11.82	17.46	23.09	28.72	34.35
81	5.07	10.70	16.33	21.96	27.59	33.22
83	3.94	9.57	15.20	20.83	26.47	32.10
85	2.82	8.45	14.08	19.71	25.34	30.97
87	1.69	7.32	12.95	18.58	24.21	29.84
89	0.56	6.19	11.82	17.46	23.09	28.72
91		5.07	10.70	16.33	21.96	27.59
93		3.94	9.57	15.20	20.83	26.47
95		2.82	8.45	14.08	19.71	25.34
97		1.69	7.32	12.95	18.58	24.21
99		0.56	6.19	11.82	17.46	23.09
101			5.07	10.70	16.33	21.96
103			3.94	9.57	15.20	20.83
105			2.82	8.45	14.08	19.71

RWCU and SFC						
RPCCW	Heat Removal Capacity (MBtu/hr)					
Tci (°F)	Rx = 150°F	Rx = 160°F	Rx = 170°F	Rx = 180°F	Rx = 190°F	Rx = 200°F
75	42.23	47.86	53.49	59.12	64.76	70.39
77	41.11	46.74	52.37	58.00	63.63	69.26
79	39.98	45.61	51.24	56.87	62.50	68.13
81	38.85	44.48	50.12	55.75	61.38	67.01
83	37.73	43.36	48.99	54.62	60.25	65.88
85	36.60	42.23	47.86	53.49	59.12	64.76
87	35.47	41.11	46.74	52.37	58.00	63.63
89	34.35	39.98	45.61	51.24	56.87	62.50
91	33.22	38.85	44.48	50.12	55.75	61.38
93	32.10	37.73	43.36	48.99	54.62	60.25
95	30.97	36.60	42.23	47.86	53.49	59.12
97	29.84	35.47	41.11	46.74	52.37	58.00
99	28.72	34.35	39.98	45.61	51.24	56.87
101	27.59	33.22	38.85	44.48	50.12	55.75
103	26.47	32.10	37.73	43.36	48.99	54.62
105	25.34	30.97	36.60	42.23	47.86	53.49

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2 ECCS-Shutdown

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ECCS injection/spray subsystem inoperable.	A.1 Restore required ECCS injection/spray subsystem to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
C. Two required ECCS injection/spray subsystems inoperable.	C.1 Initiate action to suspend OPDRVs.	Immediately
	<u>AND</u>	
	C.2 Restore one ECCS injection/spray subsystem to OPERABLE status.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action C.2 and associated Completion Time not met.	D.1 Initiate action to restore primary containment to OPERABLE status.	Immediately
	<u>AND</u>	
	D.2 Initiate action to isolate required primary containment penetration flow paths.	Immediately
	<u>AND</u>	
	D.3 ------------NOTE------------ Entry and exit is permissible under administrative control. --------------------------------- Initiate action to close one door in each primary containment air lock.	Immediately

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 -----------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	 12 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

QUESTION 86 HANDOUT

INITIATING CONDITION MATRIX

RECOGNITION CATEGORY		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
Abnormal Rad Levels / Radiological Effluent	Radiological Effluents	**AG1** Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1000 mR TEDE or 5000 mR thyroid CDE for the actual or projected duration of the release using actual meteorology *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**AS1** Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 100 mR TEDE or 500 mR thyroid CDE for the actual or projected duration of the release *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**AA1** Any release of gaseous or liquid radioactivity to the environment > 200 times the ODCM limit for ≥ 15 minutes *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**AU1** Any release of gaseous or liquid radioactivity to the environment > 2 times the ODCM limit for ≥ 60 minutes *Op Mode: 1, 2, 3, 4, 5, DEFUELED*
	Abnormal Rad Levels			**AA2** Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**AU2** UNPLANNED rise in plant radiation levels *Op Mode: 1, 2, 3, 4, 5, DEFUELED*
				AA3 Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	

INITIATING CONDITION MATRIX

RECOGNITION CATEGORY		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
Fission Product Barrier Degradation	**FPB Loss / Potential Loss**	**FG1** Loss of ANY two barriers AND loss or potential loss of the third barrier. *Op Mode: 1, 2, 3*	**FS1** Loss or potential loss of ANY two barriers *Op Mode: 1, 2, 3*	**FA1** ANY loss or ANY potential loss of EITHER fuel clad or RCS *Op Mode: 1, 2, 3*	**FU1** ANY loss or ANY potential loss of containment *Op Mode: 1, 2, 3*
Hazards and Other Conditions Affecting Plant Safety	**Security Events**	**HG1** HOSTILE ACTION resulting in loss of physical control of the facility *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HS1** HOSTILE ACTION within the PROTECTED AREA *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HU1** Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant *Op Mode: 1, 2, 3, 4, 5, DEFUELED*

INITIATING CONDITION MATRIX

RECOGNITION CATEGORY		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
Hazards and Other Conditions Affecting Plant Safety	**Discretionary**	**HG2** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HS2** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HA2** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT. *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HU2** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE *Op Mode: 1, 2, 3, 4, 5, DEFUELED*
	Control Room Evacuation		**HS3** Control Room evacuation has been initiated and plant control cannot be established *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HA3** Control Room evacuation has been initiated *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	
	Fire			**HA4** FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HU4** FIRE within PROTECTED AREA boundary not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA *Op Mode: 1, 2, 3, 4, 5, DEFUELED*

INITIATING CONDITION MATRIX

RECOGNITION CATEGORY		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
Hazards and Other Conditions Affecting Plant Safety	Toxic or Flammable gases			**HA5** Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HU5** Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS *Op Mode: 1, 2, 3, 4, 5, DEFUELED*
	Natural Phenomena			**HA6** Natural or destructive phenomena affecting VITAL AREAS *Op Mode: 1, 2, 3, 4, 5, DEFUELED*	**HU6** Natural or destructive phenomena affecting the PROTECTED AREA *Op Mode: 1, 2, 3, 4, 5, DEFUELED*
System Malfunction	Loss of AC Power	**SG1** Prolonged loss of all offsite and all onsite AC power to emergency busses *Op Mode: 1, 2, 3*	**SS1** Loss of all offsite and all onsite AC power to emergency busses for ≥ 15 minutes *Op Mode: 1, 2, 3*	**SA1** AC power capability to emergency busses reduced to a single power source for ≥ 15 minutes such that any additional single failure would result in station blackout *Op Mode: 1, 2, 3*	**SU1** Loss of all offsite AC power to emergency busses for ≥ 15 minutes *Op Mode: 1, 2, 3*

INITIATING CONDITION MATRIX

RECOGNITION CATEGORY		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
System Malfunction	**Failure of Reactor Protection System**	**SG3** Automatic scram and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists *Op Mode: 1, 2*	**SS3** Automatic scram fails to shutdown the reactor and the manual actions taken from the reactor control console are not successful in shutting down the reactor *Op Mode: 1, 2*	**SA3** Automatic scram fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor *Op Mode: 1, 2*	
	Loss of DC Power		**SS4** Loss of all vital DC power for ≥ 15 minutes *Op Mode: 1, 2, 3*		

INITIATING CONDITION MATRIX

RECOGNITION CATEGORY		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
System Malfunction	**Loss of Annunciators / Indication**		**SS6** Inability to monitor a SIGNIFICANT TRANSIENT in progress *Op Mode: 1, 2, 3*	**SA6** UNPLANNED loss of safety system annunciation or indication in the control room with either (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory non-alarming indicators are not available *Op Mode: 1, 2, 3*	**SU6** UNPLANNED loss of safety system annunciation or indication in the Control Room for ≥ 15 minutes *Op Mode: 1, 2, 3*
	RCS Leakage				**SU7** RCS leakage *Op Mode: 1, 2, 3*
	Los of Communication				**SU8** Loss of all onsite or offsite communications capabilities. *Op Mode: 1, 2, 3*
	Cladding Degradation				**SU9** Fuel clad degradation *Op Mode: 1, 2, 3*
	Inadvertent Criticality				**SU10** Inadvertent criticality *Op Mode: 3*

INITIATING CONDITION MATRIX

RECOGNITION CATEGORY		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
System Malfunction	**TS LCO Limit Exceeded**				**SU11** Inability to reach required operating mode within Technical Specification limits *Op Mode: 1, 2, 3*
Cold Shutdown / Refueling	**RCS Leakage**	**CG1** Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged *Op Mode:4, 5*	**CS1** Loss of RCS/RPV inventory affecting core decay heat removal capability *Op Mode:4, 5*	**CA1** Loss of RCS/RPV inventory *Op Mode:4, 5*	**CU1** RCS leakage *Op Mode:4*
	Loss of RCS Inventory				**CU2** UNPLANNED loss of RCS/RPV inventory *Op Mode: 5*
	Loss of Decay Heat Removal			**CA3** Inability to maintain plant in cold shutdown *Op Mode:4, 5*	**CU3** UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV *Op Mode: 4, 5*
	Loss of AC Power			**CA5** Loss of all offsite and all onsite AC power to emergency busses for ≥ 15 minutes *Op Mode: 4, 5, Defueled*	**CU5** AC power capability to emergency busses reduced to a single power source for ≥ 15 minutes such that any additional single failure would result in station blackout *Op Mode: 4, 5*

INITIATING CONDITION MATRIX

RECOGNITION CATEGORY		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
Cold Shutdown / Refueling	**Loss of DC Power**				**CU6** Loss of required DC power for ≥ 15 minutes *Op Mode: 4, 5*
	Inadvertent Criticality				**CU7** Inadvertent criticality *Op Mode: 4, 5*
	Loss of Communication				**CU8** Loss of all onsite or offsite communications capabilities *Op Mode: 4, 5, Defueled*
ISFSI	**Confinement Boundary Damage**				**E-HU1** Damage to a loaded cask CONFINEMENT BOUNDARY *Op Mode: All*

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENT

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
Radiological Effluent	**AG1** 1 2 3 4 5 D Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1000 mR TEDE or 5000 mR thyroid CDE for the actual or projected duration of the release using actual meteorology **Emergency Action Level(s):** (1 or 2 or 3) ***NOTE:*** *The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.* 1. VALID reading on any of the radiation monitors in Table R1 > the GENERAL EMERGENCY reading for ≥ 15 minutes **OR** 2. Dose assessment using actual meteorology indicates doses > 1000 mR TEDE or 5000 mR thyroid CDE at or beyond the SITE BOUNDARY **OR** 3. Field survey results indicate closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate thyroid CDE > 5000 mR for one hour of inhalation, at or beyond the SITE BOUNDARY	**AS1** 1 2 3 4 5 D Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 100 mR TEDE or 500 mR thyroid CDE for the actual or projected duration of the release **Emergency Action Level(s)** (1 or 2 or 3) ***NOTE:*** *The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, the classification should be based on EAL #2 instead of EAL #1. Do not delay declaration awaiting dose assessment results.* 1. VALID reading on any of the radiation monitors in Table R1 > the SITE AREA EMERGENCY reading for ≥ 15 minutes **OR** 2. Dose assessment using actual meteorology indicates doses > 100 mR TEDE or 500 mR thyroid CDE at or beyond the SITE BOUNDARY **OR** 3. Field survey results indicate closed window dose rates > 100 mR/hr expected to continue for ≥ 60 minutes; or analyses of field survey samples indicate thyroid CDE > 500 mR for one hour of inhalation, at or beyond the SITE BOUNDARY	**AA1** 1 2 3 4 5 D Any release of gaseous or liquid radioactivity to the environment > 200 times the ODCM limit for ≥ 15 minutes **Emergency Action Level(s)**: (1 or 2 or 3) ***NOTE:*** *The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.* 1. VALID reading on any of the radiation monitors in Table R1 > the ALERT reading for ≥ 15 minutes **OR** 2. For RMS-RE107 effluent monitor: **EITHER** VALID reading > 200 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 15 minutes **OR** VALID reading > 1.27E-01 µCi/ml for ≥ 15 minutes **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 times the ODCM limit for ≥ 15 minutes	**AU1** 1 2 3 4 5 D Any release of gaseous or liquid radioactivity to the environment > 2 times the ODCM limit for ≥ 60 minutes **Emergency Action Level(s):** (1 or 2 or 3) ***NOTE:*** *The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.* 1. VALID reading on any of the radiation monitors in Table R1 > the NOUE reading for ≥ 60 minutes **OR** 2. VALID reading on RMS-RE107 effluent monitor > 2 times the alarm setpoint established by a current radioactivity discharge permit for ≥ 60 minutes **OR** 3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 times the ODCM limit for ≥ 60 minutes

Table R1 EAL THRESHOLD

Method	GENERAL DRMS Threshold	SITE AREA DRMS Threshold	ALERT DRMS Threshold	NOUE DRMS Threshold
Main Plant Vent Primary Secondary	4GE125 4.50E+08 µCi/sec N/A	4GE125 4.50E+07 µCi/sec N/A	4GE125 3.06E+07 µCi/sec 1GE126 2.82E-01 µCi/ml	4GE125 3.06E+05 µCi/sec 1GE126 5.26E-03 µCi/ml
Fuel Building Vent Primary Secondary	4GE005 1.00E+09 µCi/sec N/A	4GE005 1.00E+08 µCi/sec N/A	4GE005 2.19E+06 µCi/sec 5GE005 2.82E-01 µCi/ml	4GE005 2.19E+04 µCi/sec 5GE005 4.65E-03 µCi/ml
Radwaste Building Vent Primary Secondary	N/A	N/A	4GE006 2.58E+06 µCi/sec 5GE006 6.84E-02 µCi/ml	4GE006 2.58E+04 µCi/sec 5GE006 6.84E-04 µCi/ml

Plant Modes (white boxes indicate applicable modes) 1 Power Operations 2 Startup 3 Hot Shutdown 4 Cold Shutdown 5 Refuel D Defueled

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENT

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
Abnormal Radiation Levels			**AA2** 1 2 3 4 5 D Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel **Emergency Action Level(s)**: (1 or 2) 1. A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered **OR** 2. A VALID reading on any of the following radiation monitors due to damage to irradiated fuel or loss of water level: RMS-RE140 2000 mR/hr RMS-RE141 2000 mR/hr RMS-RE192 2000 mR/hr RMS-RE193 2000 mR/hr RMS-RE5A 1.64E+03 µCi/sec RMS-RE5B (GE) 5.29E-04 µCi/ml	**AU2** 1 2 3 4 5 D UNPLANNED rise in plant radiation levels **Emergency Action Level(s)**: (1 or 2) 1. a. UNPLANNED water level drop in a reactor refueling pathway as indicated by any of the following: a. Water level drop in the reactor refueling cavity, spent fuel pool, or fuel transfer canal indication on Control Room Panel 870 b. Personnel observation by visual or remote means. **AND** b UNPLANNED VALID area radiation monitor alarm on any of the following: RMS-RE140 RMS-RE141 RMS-RE192 RMS-RE193 **OR** 2. UNPLANNED VALID area radiation monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels ***NOTE**: For area radiation monitors with ranges incapable of measuring 1000 times normal* levels, classification shall be based on VALID full scale indications unless surveys confirm that area radiation levels are below 1000 times normal* within 15 minutes of the area radiation monitor indications going full scale.* *Normal can be considered the highest reading in the past 24 hours excluding the current peak value.
			AA3 1 2 3 4 5 D Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions **Emergency Action Level(s)**: 1. Dose rate > 15 mR/hr in any of the following areas requiring continuous occupancy to maintain plant safety functions: Main Control Room CAS	

Plant Modes (white boxes indicate applicable modes) 1 Power Operations 2 Startup 3 Hot Shutdown 4 Cold Shutdown 5 Refuel D Defueled

FISSION PRODUCT BARRIER

FPB Loss / Potential Loss	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOUE
	FG1 1 2 3 4 5 D Loss of ANY two barriers AND loss or potential loss of the third barrier **Emergency Action Level(s):** 1. Loss of any two barriers **AND** Loss or potential loss of the third barrier	**FS1** 1 2 3 4 5 D Loss or potential loss of ANY two barriers **Emergency Action Level(s):** 1. Loss or potential loss of any two barriers	**FA1** 1 2 3 4 5 D ANY loss or ANY potential loss of EITHER fuel clad or RCS **Emergency Action Level(s):** 1. Any loss or any potential loss of fuel clad **OR** Any loss or any potential loss of RCS	**FU1** 1 2 3 4 5 D ANY loss or ANY potential loss of containment **Emergency Action Level(s):** 1. Any loss or any potential loss of containment

FUEL CLAD (FC) Barrier			REACTOR COOLANT SYSTEM (RC) Barrier			PRIMARY CONTAINMENT (PC) Barrier		
Parameter	Loss	Potential Loss	Parameter	Loss	Potential Loss	Parameter	Loss	Potential Loss
FC1 Primary coolant activity level	Coolant activity > 300 µCi/gm dose equivalent I-131	None	RC1 Drywell pressure	Drywell pressure > 1.68 psid with indications of reactor coolant leak in drywell	None	PC1 Primary containment conditions	1. Rapid unexplained loss of PC pressure following initial pressure rise **OR** 2. PC pressure response not consistent with LOCA conditions	1. PC pressure > 15 psig and rising **OR** 2. a. PC hydrogen in the unsafe zone of HDOL curve **OR** b. DW hydrogen concentration > 9% **OR** 3. RPV pressure and suppression pool temperature cannot be maintained below the HCTL
FC2 Reactor vessel water level	RPV water level cannot be restored and maintained above -187 inches	RPV water level cannot be restored and maintained above -162 inches or cannot be determined	RC2 Reactor vessel water level	RPV water level cannot be restored and maintained above -162 inches or cannot be determined	None	PC2 Reactor vessel water level	None	Entry into PC flooding procedures SAP-1 and SAP-2
FC3 Primary Containment radiation monitors	Containment radiation monitor RMS-RE16 reading > 3,000 R/hr	None	RC3 RCS Leak Rate	1. UNISOLABLE main steam line break as indicated by the failure of both MSIVs in any one line to close **AND** High MSL flow annunciator (P601-19A- A2) **OR** 2. Indication of an UNISOLABLE HPCS, feedwater, RWCU or RCIC break **OR** 3. Emergency RPV depressurization is required	1. RCS leakage > 50 gpm inside the drywell **OR** 2. UNISOLABLE RCS leakage outside PC as indicated by exceeding either of the following: a. Max Normal Operating Temperature (Table F2) **OR** b. Max Normal Area Radiation (Table F2)	PC3 Primary containment isolation failure or bypass	1. a. Failure of all valves in any one line to close **AND** b. Direct downstream pathway to the environment exists after PC isolation signal **OR** 2. Intentional PC venting per EOPs or SAPs **OR** 3. UNISOLABLE RCS leakage outside PC as indicated by exceeding either of the following: a. Max Safe Operating Temperature (Table F1) **OR** b. Max Safe Area Radiation (Table F1)	None
			RC4 Drywell radiation	Drywell radiation monitor RMS-RE20 reading > 100 R/hr due to reactor coolant leakage	None	PC4 Primary containment radiation monitors	None	Containment radiation monitor RMS-RE16 reading > 10,000 R/hr
FC4 Emergency Director judgment	Any condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier	RC5 Emergency Director judgment	Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	PC5 Emergency Director judgment	Any condition in the opinion of the Emergency Director that indicates loss of the Primary Containment barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the Primary Containment barrier

Plant Modes (white boxes indicate applicable modes) 1 Power Operations 2 Startup 3 Hot Shutdown 4 Cold Shutdown 5 Refuel D Defueled

FISSION PRODUCT BARRIER

TABLE F1 PC 3 Loss of Primary Containment			
Parameter	**Area Temperature** Max Safe Operating Value	**Area Radiation Level** DRMS Grid 2	Max Safe Operating Value
RHR A equipment area	200º F	1213	9.5E+03 mR/hr
RHR B equipment area	200º F	1214	9.5E+03 mR/hr
RHR C equipment area	N/A	1215	9.5E+03 mR/hr
RCIC room	200º F	1219	9.5E+03 mR/hr
MSL Tunnel	200º F	N/A	
RWCU pump room 1 (A) / 2 (B)	200º F	N/A	

TABLE F2 RC 3 Potential Loss of RCS			
Parameter	**Area Temperature** (isolation temperature alarm)	**Area Radiation Level** DRMS Grid 2	Max Normal Operating Value
RHR A equipment area	117º F (P601-20A-B4)	1213	8.2E+01 mR/hr
RHR B equipment area	117º F (P601-20A-B4)	1214	8.2E+01 mR/hr
RHR C equipment area	N/A	1215	8.2E+01 mR/hr
RCIC room	182º F (P601-21A-B6)	1219	1.20E+02 mR/hr
MSL Tunnel	173º F (P601-19A-A1/A3/B1/B3)	N/A	
RWCU pump room 1 (A) / 2 (B)	165º F (P680-1A-A2/B2)	N/A	