RO

Answer Sheet

Exam Title: 2014 ILT SRO NRC Written Exam

Name KEY

Date

SRO only key

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

Deleted

B or D Correct

U.S. Nuclear Regulatory Commission

Site-Specific RO Written Examination KEY

Applicant Information

☐ Name:

Date: 08-04-2014	Facility/Unit: Cooper Nuclear 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	~~75~~ 74 Points
Applicant's Score	_____ Points
Applicant's Grade	_____ Percent

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295001.AK2.06	
Importance Rating	3.8	

295001 Partial or Complete Loss of Forced Core Flow Circulation

K2.Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following:

AK2.06 Reactor power

Question: 1

With the plant operating at 98% power during a coast down (Rods Full Out, Total Core Flow 73.5 Mlb/hr). Both RRMG sets are operating at 100% rated speed when RRMG set A begins to slowly lower recirculation flow. The following conditions exist:

- Reactor power is 85% and lowering.
- RRMG A SCOOPTUBE LOCKOUT button is depressed.
- RRMG A continues to slowly lower recirculation flow.
- Currently Reactor power is 65%, and Total Core Flow is 31 Mlb/hr.
- Gardel Stability evaluation report runs.

What action is required?

a. Trip RRMG A to preclude RR pump vibration.

b. Insert Control Rods to exit the Stability Exclusion Region.

c. Lower RRMG B speed to minimize the recirculation flow mismatch.

d. Raise the speed of RR Pump B to exit the Stability Exclusion Region.

Answer:

b. Insert Control Rods to exit the Stability Exclusion Region.

Explanation (Optional):

Operation is in the stability exclusion Region of the power-to-flow map. 2.4RR Attachment 3 is entered for operation in this region. Technical Specifications LCO 3.4.1, Condition A requires immediate action to be initiated to exit this operating region. 2.4RR, Attachment 3 directs exiting the region by raising RR flow or inserting control rods. Raising the speed of the 1B pump cannot be performed because it is at maximum flow. A Caution in 2.4RR, Attachment 3 warns that

operating RRMG set speeds above 100% can cause vibration induced damage to internal RPV components and large pipes. Also, because RRMG A will not lock up, manual control is not allowed. The only other option is to insert the Emergency Power Reduction Rods per Procedure10.13.

Distracters:

a. is incorrect because RR pump vibration would not be an issue. Flow mismatches cause jet pump vibration issues but not RR pump vibration issues. The candidate that does not understand what components could be susceptible to vibration issues with flows mismatched would choose this answer.

c. is incorrect because this would put operation further into the stability exclusion region. This action would be appropriate for operation outside the stability exclusion region. The candidate that believes that the RR pump speed mismatch is the major concern would choose this answer.

d. is incorrect because high vibration and damage to reactor internal components, large and small piping, and pipe-mounted equipment may occur. This is stated in a Caution Statement in 2.4RR Attachment 3. The candidate that believes exiting the stability exclusion area by raising RR flow because it is faster than inserting control rods would choose this answer if they forgot about the caution.

Technical Reference(s):
Procedure 2.4RR, Rev 40
Procedure 2.1.10, Rev 106

Proposed references to be provided to applicants during examination: Power to Flow Map in 2.1.10

Learning Objective:
Given plant condition(s) and the applicable Abnormal / Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source: Bank # 15414
Modified Bank #
New

Question History: Last NRC Exam 2008 NRC

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

10 CFR Part 55 Content: 55.41 7
55.43

Difficulty: 4

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295003.AK3.03	
Importance Rating	3.5	

295003 Partial or Complete Loss of AC

K3. Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER:

AK3.03 Load Shedding

Question: 2

What is the primary reason per USAR for load shedding buses 1F and 1G during a loss of off-site power?

a. To protect 4160V buses 1F and 1G from degraded voltage.

b. To inhibit the automatic closure of breakers 1FS (1GS) on low voltage.

c. To prevent an immediate diesel overload when they tie onto their buses.

d. To prevent re-closure of breakers 1FE (1GE) due to the anti-pumping feature.

Answer:

c. To prevent an immediate diesel overload when they tie onto their buses.

Explanation (Optional):

From Student Text COR001-01-02 Page 39 of 147

To prevent an immediate overloading of the diesels following an automatic start, a condition called load shedding occurs prior to output circuit breaker closure. On load shedding, all the 4160V breakers, except breakers 1FE, 1GE, SS1F, and SS1G on the 4160V buses 1F and 1G will trip on under voltage. Emergency systems or normal equipment are then restarted automatically in a timed sequence to prevent overloading the power supply, dropping the supply voltage below acceptable limits.

Distracters:

a. This is the function of the degraded voltage scheme as described here: Relay 27X15/1F(1G) - This relay energizes when a degraded voltage condition has been sensed on the associated emergency bus for 7.5 seconds, as a result of the energization

of relay 27X17/1F(1G). When relay 27X15/1F(1G) picks up, it trips breaker 1FS(1GS). (This protection exists to protect the emergency buses from degraded voltage while being supplied from the Emergency Transformer.) The candidate that does not understand the reason for load shedding protecting the DGs from overloading would choose this answer.

b. This is the function of the degraded voltage scheme as described here: This relay energizes when Emergency Transformer secondary under voltage (<4330 V) is sensed. This relay, when energized, inhibits the automatic closure of breaker 1FS (1GS). (These relays are fed from relays 27/ET3 and 27/ET4, which drop out on low voltage of <4330 V on the secondary side of the Emergency Transformer following a 7.5 second time delay. They will reset immediately when voltage is restored above 4330 V.) The candidate that believes the reason for load shedding is the same as degraded voltage would choose this answer.

d. This is the function of the anti-pumping circuit not load shedding: If breaker 1FE and/or 1GE trip <u>and</u> the 69 KV supply to the Emergency Transformer is deenergized, the 1FE and/or 1GE breakers <u>cannot</u> be reclosed due to the anti-pumping feature of the breakers (there is an auto closure signal present from the under voltage relays 27X/ET1 and 27XET2). The anti-pumping feature can be reset by momentarily operating the racking screw shutter lever. When the shutter is opened, then closed, the breaker will also close. The candidate that believes the reason for the anti-pumping feature is the same as load shedding would choose this answer.

Technical Reference(s):
Student Text COR0010101, Rev 45

Proposed references to be provided to applicants during examination: None

Learning Objective:

Briefly describe the following concepts as they apply to AC Electrical Distribution System: Load shedding

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	55.41 5
	55.43

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295004.AA2.02	
Importance Rating	3.5	

295004 Partial or Total Loss of DC Pwr

AA2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:

AA2.02 Extent of partial or complete loss of D.C. power

Question: 3

The plant is operating at power when the following significant conditions are noted by the crew:

- 9-3-1/B-7, CORE SPRAY A LOGIC POWER FAILURE annunciates.
- 9-4-1/A-3, RCIC LOGIC POWER FAILURE annunciates.
- Indicating lights for 4160VAC buses A, C, and E are <u>ON</u>**.**

What is the extent of the failure?

a. Only Panel AA1 is lost.

b. Only Panel AA2 is lost.

c. Only Panel AA1 and Panel AA2 are lost.

d. 125 VDC Distribution Panel A is lost.

Answer:

b. Only Panel AA2 is lost.

Explanation (Optional):

Core Spray A logic is powered from 125VDC Panel AA-2, Circuit 6. RCIC logic is powered from 125VDC Panel AA-2, Circuit 11. As a result, it is known that all of Panel AA-2 is without power. Control power for 4160VAC buses A, C, and E is determined by observing Breaker 1AS, 1AN, 1AE and 1CN indications on the control room panels. If an indicating light is ON for the given breaker position, it can be deduced those breaker control circuits have power. The power supply to these breaker control circuits is 125VDC Panel AA-1.

Distracters:

a. is incorrect because loss of AA-1 would not affect RCIC or Core Spray A logic. Panel AA-1 does supply logic to some system breakers so this answer is plausible. The

candidate that does not realize the logic power supply for Core Spray and RCIC would choose this answer.

c. is incorrect because no indication of a loss of AA-1 is present and if AA-1 were lost control power to 4160 A,C and E would be lost. Panel AA-1 does supply logic to some system breakers so this answer is plausible. That candidate that does not realize the logic power supply to the given 4160VAC bus logic would choose this answer..

d. is incorrect because Panel AA-1 still has power. The loss of Distribution Panel A would cause a loss of power to panel AA-2 so this answer is plausible. The candidate that does not know the general 125VDC distribution layout would choose this answer.

Technical Reference(s):
Emergency Procedure 5.3DC125, Rev 31

Proposed references to be provided to applicants during examination: None

Learning Objective:

Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Components using DC control power (i.e., breakers)

Question Source: Bank # 23416
Modified Bank #
New

Question History: Last NRC Exam 2006 NRC

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

10 CFR Part 55 Content: 55.41 10
55.43

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295005.AA1.01	
Importance Rating	3.1	

295005 Main Turbine Generator Trip

AA1. Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP:

AA1.01 Recirculation system: Plant Specific

Question: 4

The reactor is operating at rated power with the following 4160V breaker lineup:

- 1CN closed
- 1DS closed

The main generator trips on load reject.

What is the status of the Reactor Recirculation (RR) system two minutes later?

a. Only RRMG B is operating and its field breaker is open.

b. Only RRMG B is operating and its field breaker is closed.

c. Both RRMGs are tripped and their field breakers are open.

d. Both RRMGs are operating and their field breakers are closed.

Answer:

a. Only RRMG B is operating and its field breaker is open.

Explanation (Optional):

A main turbine trip from 100% power results in reactor pressure rising above the RPT pressure trip set point (≤ 1072 psig) due to the turbine stop valve fast closure. The high pressure RPT trip opens both RR pump field breakers. The main generator trip causes a loss of power to the Normal transformer. The 4160V switchgear 1C and 1D is no longer powered from the Normal Transformer but the Startup Station Transformer power is available to the switchgear. As a result, the breaker that is closed (1DS) remains closed and the RRMG continues to run even though the field breaker is open. Breaker 1CN trips open so RRMG A trips. Bounded the stem with 2 minutes to disqualify the high MG Set temperature trips.

Distracters:

b. is incorrect because RRMG set B continues to run (1DS closed) and powered from the Startup Station transformer. Its field breaker is open because of the RPT trip. The candidate that fails to realize the RPT trip occurs on a turbine trip from full power may select this answer. This answer is plausible because both the field breakers don't always trip on a reactor scram.

c. is incorrect because RRMG set B continues to run (1DS closed) and powered from the Startup Station transformer. The candidate that believes the RRMG drive motor breakers fast transfer on a loss of the Normal may select this answer. This answer is plausible the field breakers trip on a RPT condition.

d. is incorrect because RRMG set B continues to run (1DS closed) and powered from the Startup Station transformer and both field breakers are open because of the RPT trip. The candidate that does not realize the RPT trip occurs on a turbine trip from full power may select this answer. This answer is plausible because both RRMGs can remain operating post scram if they are both powered from the Startup Station Transformer.

Technical Reference(s):
Student Text COR0022202, Rev 31

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022202001040E Describe the interrelationships between Reactor Recirculation system or the Recirculation Flow Control system and the following: Main Turbine Generator Trip

Question Source: Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content: 55.41 7
55.43

Difficulty 4

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A # 295006	AK1.01	
Importance Rating	3.7	

295006 Scram

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

AK1.01 Decay heat generation and removal

Question: 5

With the plant operating at rated power after 100 continuous days of operation, the following occurs:

- 0800 A loss of condenser vacuum begins.
- 0801 Reactor power is lowered 50%.
- 0802 A manual scram is inserted with all control rods fully inserting.
- 0803 Condenser vacuum is 6" Hg and steady.

What method(s) of decay heat removal is/are appropriate to utilize over the next 30 minutes?

a. One RFP turbine only.

b. The RCIC turbine only.

c. HPCI initially and then RCIC.

d. The MT BPVs initially and then one RFP turbine.

Answer:

c. HPCI initially and then RCIC.

Explanation (Why distractors are incorrect and why correct answer is correct):
Decay heat produces approximately 7% power 10 seconds after a scram and drops to approximately 3% after a couple of minutes. Initially the decay heat removal is handled by HPCI but after a few minutes (approximately 15 minutes), RCIC is able to handle the decay heat. With condenser vacuum at 6" Hg, the MSIVs, main turbine bypass valves, and RFPTs are no longer available to respond to decay heat generation. This leaves systems upstream of the MSIVs. Because a rapid power reduction was made before the manual scram was initiated, the SRVs are not required to respond when the MSIVs closed due to loss of condenser vacuum. the generator is not rate limited on down power evolutions.

Distracters:

a. This answer is incorrect because the RFPTs are tripped on low condenser vacuum and the MSIVs are closed. The candidate that does not recall the low condenser vacuum trip setting of the RFPTs may select this answer as the RFPTs are normally available post scram. This answer is plausible because one RFPT can handle the decay heat generation rate post scram.

b. This answer is incorrect because RCIC is unable to handle the decay heat generation rate for the first 15 minutes after the scram. It will need assistance. In this case HPCI will be required initially. The candidate that does not recall the capability of RCIC post scram may select this answer. This answer is plausible because RCIC is normally the post scram decay heat removal system but only later after HPCI initially is used.

d. This answer is incorrect because the Main Turbine Bypass Valves and the RFPT are not available for use with condenser vacuum at this level (< 7" Hg required for both). The main turbine bypass valves are normally the post scram decay heat removal system and the candidate that does not recall their condenser vacuum interlock may select this answer. This answer is plausible because the main turbine bypass valves and the RFP turbine is the normal post scram pressure control method.

Technical Reference(s):
DCD 18 Dated 2/2/09 (See attached partial)

Proposed references to be provided to applicants during examination: None

Learning Objective(s):
COR0021102001110A, State the reason(s) for the following: Use of HPCI for pressure control
COR0021802001110A, State the reason for the following: Use of RCIC for pressure and level control

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

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

10 CFR Part 55 Content: 41.8

Difficulty 3

Design Criteria Document 18

REACTOR CORE ISOLATION COOLING SYSTEM DESIGN CRITERIA, REQUIREMENTS AND DESIGN BASIS

Starting with the system level design criteria, derived from the highest tier design input documents, including the Nuclear Safety Operational Analysis (NSOA), a set of system level design requirements was developed for each system design criterion. These system design requirements are more specific than the design criteria, and constitute the specific functions and parameters that must be met in order to meet the criterion. For each system requirement, a set of components and structures was identified. For each structure or component, a set of functional and design parameters was identified which are essential to meeting the system level design requirements.

10CFR50.2 defines the "design basis" as: "that information which identifies the specific functions to be performed by a structure, system or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design." In the following sections, the design basis for the RCIC System is documented. Each system functional criterion is followed by the design requirements that implement the criterion and the basis is documented for each design criterion and design requirement.

System Design Criterion 1 - RCIC System Cooling Capability

The RCIC System shall provide coolant makeup to the vessel to keep the core covered and cooled for reactor isolation accompanied by a loss of feedwater flow and shall be able to depressurize the reactor so that shutdown cooling systems may be placed into operation.

Basis - The General Electric Design Specification[1], NEDE-10370[2], and USAR[3] state that the system shall restore and maintain the water level in the reactor vessel after a reactor isolation and loss of feedwater flow to prevent reactor fuel from overheating.

System Design Requirement 1 - RCIC Pump Performance

The RCIC pump shall be capable of delivering flow to the reactor vessel to maintain the vessel water level.

Basis - The General Electric Design Specification[4] states that the RCIC capacity shall be sufficient to prevent the reactor vessel water level from dropping to the top of the core. The RCIC flow rate is approximately equal to the reactor water boil off rate 15 minutes after shutdown.

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1 General Electric Design Specification 22A1354, Rev, 4, 4/29/71, Reactor Core Isolation Cooling System, Section 3.2

2 General Electric NEDE-10370, Design Safety Standards for Boiling Water Reactors, 1974, Section 6.2.13.1

3 Updated Safety Analysis Report, Volume VII Appendix G, Sections 6.3 and 6.4

4 General Electric Design Specification 22A1354, Rev. 4, 4/29/71, Reactor Core Isolation Cooling System, Section 4.1.2

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A # 295016	2.2.44	
Importance Rating	4.2	

295016 Control Room Abandonment

AK 1 Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM ABANDONMENT:

2.2.44 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

Question: 6

The control room is recovering from a shutdown from outside the control room. RHR loop B is delivering 5000 gpm flow on RHR-FI-1133B in torus cooling. The control room operator has performed the appropriate procedure steps and is ready for the control of RHR transfer back to the control room. All RHR ISOLATION switches on the ASD Panel are in NORM.

How does Procedure 5.1ASD, ALTERNATE SHUTDOWN, direct the control room operator to verify operation of RHR is transferred from the ASD Panel to Panel 9-3-1?

a. Verify RHR Pump B indicating lights have red light on and green light off.

b. Verify RHR Pump D indicating lights have red light on and green light off.

c. Throttle RHR-MO-34B, SUPPR POOL COOLING INBD THROTTLE VLV, open until a rise in flow on RHR-FI-133B is observed.

d. Throttle RHR-MO-34B, SUPPR POOL COOLING INBD THROTTLE VLV, closed until a drop in flow on RHR-FI-133B is observed.

Answer:

d. Throttle RHR-MO-34B, SUPPR POOL COOLING INBD THROTTLE VLV, closed until a drop in flow on RHR-FI-133B is observed.

Explanation (Why distractors are incorrect and why correct answer is correct):
Procedure 5.1ASD, Step 11.2.4 directs throttling RHR-MO-34B closed until a drop in RHR flow is observed on RHR-FI-133B.

Distracters:

a. This answer is incorrect because the RHR pump control is not transferred to the ASD panel. RHR Pump B is not operating in this event. Verifying the pump is still running is not a transfer check directed by procedure 5.1ASD. The operator that does not realize the RHR pump control is not transferred may select this answer. This answer is plausible because several RHR valves have their control transferred to the ASD panel.

b. This answer is incorrect because the RHR pump control is not transferred to the ASD panel. Verifying the pump is still running is not a transfer check directed by procedure 5.1ASD. The operator that does not realize the RHR pump control is not transferred may select this answer. This answer is plausible because several RHR valves have their control transferred to the ASD panel.

c. The answer is incorrect because the procedure directs throttling closed on the valve to verify control transfer. The operator that does not know this may select this answer. This answer is plausible because the RHR-MP-34B is throttled to verify control transfer.

Technical Reference(s):
Emergency Procedure 5.1ASD, Rev 16 Alternate Shutdown

Proposed references to be provided to applicants during examination: None

Learning Objective(s):
COR0023402001020H Describe the interrelationship between ASD and the following: Residual Heat Removal (RHR) system

Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New X	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis X	
10 CFR Part 55 Content:	41.13	
Difficulty	3	

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295018.AK3.02	
Importance Rating	3.3	

295018 Partial or Total Loss of CCW

AK 3 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:

AK3.02 Reactor power reduction

Question 7

The plant is operating during the summer months. Due to equipment problems, REC heat exchanger outlet temperature is approaching 98°F. The CRS directed reactor power to be lowered because of the high river water temperatures.

Why is reactor power lowered for this condition?

a. To maintain the SW system operable.

b. To maintain the REC system operable.

c. To allow the REC TCVs to partially close.

d. To allow the SW TCVs to partially close.

Answer:

b. To maintain the REC system operable.

Explanation (Optional):

The REC system heat exchanger outlet temperature must be kept below 98°F requires entry into the appropriate Condition and Required Action of LCO 3.7.3 per a NOTE in Procedure 2.2.65.1, REC OPERATIONS (Section 6, NOTE before Step 6.5). Per TS LCO 3.7.3, the REC system operability is based on maintaining the supply water below 100°F (SR 3.7.3.2). Lowering reactor power lowers the heat load on the REC cooled equipment. A major heat load on the REC system is the drywell fan coil units due to RR pump operation. Lowering power with the RR pumps significantly lowers the heat load on the REC system. Procedure 5.2REC Step 4.9.3.1 directs lowering heat loads on the REC system.

Distracters:

a. The SW system operability is not based on the REC heat exchanger outlet temperature. The SW system operability is based on the Missouri River temperature. Lowering power to maintain SW operability is not the reason for power reduction. The candidate that does not understand this may select this answer. This answer is plausible because the SW system operability is based on a cooling medium temperature.

c. Power is not lowered to maintain the TCV position and control NPSH requirements. The REC surge tank performs this function. The TCVs in the REC system are maintained in the manual position so no TCVs change position based on component temperatures. The cooling medium (Service Water) temperature valve can be in automatic and control REC cooling water temperature. The candidate that believes REC TCVs operate in automatic may select this answer. This answer is plausible because the TEC system has TCVs that automatically reposition to maintain temperature.

d. SW TCVs are operated in automatic to maintain their component cooling within required bands. The TCVs are not designed to operate to maintain the system header pressure. While lowering reactor power will lessen the load on the SW system and allow SW TCVs to close, that is not the reason for lowering power in this instance. The candidate that does not recall that reactor power is lowered for REC operability considerations may select this answer. This answer is plausible because lowering reactor power will lower the heat load on the SW system.

Technical Reference(s):
Emergency Procedure 5.2REC, Rev 16, Loss of REC

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0021902001070A, Predict the consequences a malfunction of the following would have on the REC system: Loss of Service Water system

Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New X	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis X	
10 CFR Part 55 Content:	55.41 5	
	55.43	

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295019.AA1.04	
Importance Rating	3.3	

295019 Partial or Total Loss of Inst. Air

AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:

AA1.04 Service air isolations valves: Plant-Specific

Question: 8

The plant is operating at 100% when a medium size Service Air leak in the pipe serving the Radwaste Building occurs causing both Service Air and Instrument Air Pressure to start lowering. All available air compressors are running at full capacity.

What is the response of the Instrument and Service Air Pressure instruments in the main control room?

	SA-PI-611	**IA-PI-606**
a.	continues to lower	continues to lower
b.	lowers then recovers	continues to lower
c.	continues to lower	lowers then recovers
d.	lowers then recovers	lowers then recovers

Answer:

c. continues to lower lowers then recovers

Explanation (Optional):

The 3 station air compressors supply compressed air to the 2 station air receivers. At the outlet of the air receivers the piping separates into two main headers. One header supplies all the station Instrument Air and the other header supplies all the station Service Air. The Service Air header contains an isolation valve, SA-PCV-609, SERVICE AIR SYSTEM ISOLATION, which closes when the Service Air header pressure lowers to < 77 psig. The control room is notified when SA-PCV-609 should close by alarm A-4/B-4, SERVICE AIR ISOLATION PCV-609. The control room does not have direct PCV-609 position indication but its closure can be verified by observing Instrument Air pressure indication. As the leak depressurizes the air headers, both Service Air and Instrument Air pressures indicated in the main control room start lowering. When air receiver pressure lowers to <77 psig, SA-PCV-609, automatically closes and isolates the Service Air from the instrument air system. The leak is now isolated from Instrument Air header. The Service Air pressure indication will lower all the way to 0. The air compressors sequentially start and restore Instrument Air pressure to its normal value.

Distracters:

a. This is incorrect because when the SA-PCV-609 closes, Instrument Air recovers. A candidate might choose this answer if they fail to realize that the pressure instrument for Instrument Air will indicate the recovery when the Air leak is isolated form the Inst. Air Header.

b. This is incorrect because Service Air will not recover but Instrument Air does. A candidate might choose this answer if they confuse the Instrument Air Pressure gauge and the Service Air pressure instruments.

d. This is incorrect because Service Air will not recover but Instrument Air does. A candidate might choose this answer if they fail to realize that the air leak will completely depressurize the Service Air Header once SA-PCV-609 has closed.

Technical Reference(s):
Student Text COR0011702, Rev

Proposed references to be provided to applicants during examination: None

Learning Objective:

3. COR0011702001030A, Describe the interrelationships between the Plant Air system and the following:

   a. Service Air

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

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

10 CFR Part 55 Content: 55.41 7
55.43

Difficulty 4

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295021.AK.02	
Importance Rating	3.2	

295021 Loss of Shutdown Cooling

AK2. Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following:

AK2.02 Reactor water cleanup

Question: 9

RHR loop B is operating in the Shutdown Cooling mode of operation. Reactor water level is being maintained at 50 inches Narrow Range. A switch malfunction has caused RHR-MO-17 to close. The valve cannot be reopened and the RPV coolant starts heating up slowly.

In accordance with 2.4SDC SHUTDOWN COOLING ABNORMAL, what sub-system or system is placed in service to remove heat from the coolant?

a. RHR loop A.

b. Fuel Pool Cooling.

c. Control Rod Drive.

d. Reactor Water Clean-up.

Answer:

d. Reactor Water Clean-up.

Explanation (Optional):

The RHR-MO-17 is the Shutdown Cooling suction from the vessel and is common to both loops of RHR. When that valve closed, a limit switch causes the RHR Pumps to trip and that is an entry into the Shutdown Cooling Abnormal Procedure 2.4SDC.

From 2.4SDC, one of the Entry Conditions is an operating RHR pump trips while in SDC. And from Attachment 2 Contingency Actions for Complete Loss of SDC the last step is to place RWCU System in service per alternate heat removal section of Procedure 2.2.66.

Distracters:

a. Since RHR loop A shares a common suction through the RHR-MO-17, there is no flow path for shutdown cooling, just like loop B. A candidate who did not remember that there is a shared suction could pick this answer.

b. Since the water level in the vessel is given as 50 inches, the Fuel Pool Gates have to be installed, and therefore Fuel Pool Cooling can only supply cooling to the Spent Fuel Pool. A candidate, who did not recognize that the gates would have to be installed, could pick this answer because had the gates been removed, this would be the correct answer.

c. Since RWCU is available for Alternate Cooling, the Control Rod Drive system would not be used, because it is less effective and a feed and bleed would have to be setup with another system. A candidate, who did not realize that RWCU was available might choose this answer if they assumed that a bleed path would be set up as well. The abnormal procedure does not specify CRD be used as a heat removal system.

Technical Reference(s):
Procedure 2.2.66, Rev 103 Reactor Water Cleanup
Abnormal Procedure 2.4SDC, Rev 14, Shutdown Cooling Abnormal
Student Text, INT0320126, Rev 03

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0022302001070
INT0320126Q0Q0100

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

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

10 CFR Part 55 Content: 55.41 7
55.43

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295023.AA1.06	
Importance Rating	3.3	

295023 Refueling Accident

AA 1 Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS:

AA1.06 Neutron Monitoring

Question 10

A core reload is in progress and is almost complete. A once used fuel assembly is being lowered into the core.

Subsequently the following conditions occur:

• Continuously rising count rate is observed on all SRMs.
• REFUELING AREA HIGH RAD (9-3-1/A-10) alarms.

What does this information indicate?

a. The fuel assembly is damaged as it is lowered.

b. Reactor criticality occurs as the fuel assembly is lowered.

c. The fuel assembly is being inserted next to an SRM detector.

d. The assembly is lowered too fast and particle debris is knocked loose.

Answer:

b. Reactor criticality occurs as the fuel assembly is lowered.

Explanation (Optional):

Per Procedure 2.1.1, STARTUP, NOTE - Reactor is critical when neutron flux rises with a constant (stable) period without requiring additional control rod withdrawal. The period alarm is indicative of a large rise in neutron count rate over a short period of time. The rising count rate on the SRMs indicatives a critical reactor.

Distracters:

a. This option is incorrect because the indications are that the reactor is critical. Although a damaged fuel assembly could cause the refuel floor radiation alarm, it would not cause all the SRM indications to rise. The candidate that believes that a damaged fuel assembly would make all the SRM indications rise would choose this answer. This answer is plausible because a damaged fuel assembly would cause refueling floor radiation alarms.

c. This option is incorrect because even though inserting a fuel assembly next to an SRM could cause its indication to rise the indication would not continue to rise. Since the candidates hay have seen assemblies loaded and seen an SRM response they may choose this answer.

d. This option is incorrect because particle debris from a fuel assembly may create conditions that could cause high radiation conditions but it would not cause the SRMs to rise or to continuously rise. Only criticality would cause a continuous rise in SRM indication. A candidate who fails to analyze all the indications may choose this answer because the high radiation from a release of activated products from the fuel assembly could cause high local radiation alarms.

Technical Reference(s):

Procedure 2.1.1, Rev 171

Proposed references to be provided to applicants during examination: None

Learning Objective:

INT032010400A0600 State the conditions which indicate the Reactor is critical.

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level:

Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:

55.41 7
55.43

Difficulty: 2

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295024.G2.4.31	
Importance Rating	4.2	

295024 High Drywell Pressure

2.4.31 Knowledge of annunciator alarms, indications, or response procedures.

Question: 11

While raising power from 75% to 90% power, the following conditions exist:

- Drywell temperature has risen from 135°F to 145°F in the past 10 minutes.
- Drywell pressure is 1.5 psig and rising 0.5 psig/min.

What is/are the next appropriate operator action(s)?

a. Per Alarm 9-5-2/F-3, HIGH DRYWELL PRESSURE, scram and enter Procedure 2.1.5, REACTOR SCRAM.

b. Per Alarm 9-5-2/F-3, HIGH DRYWELL PRESSURE, ensure all available drywell FCU control switches are in OVERRIDE.

c. Per Procedure 2.4MC-RF, CONDENSATE AND FEEDWATER ABNORMAL, trip and isolate both reactor feedwater pumps.

d. Per Procedure 2.1.10, STATION POWER CHANGES, perform rapid power reduction and reduce core flow to $40x10^6$ lbs/hr.

Answer:

a. Per Alarm 9-5-2/F-3, HIGH DRYWELL PRESSURE, scram and enter Procedure 2.1.5, REACTOR SCRAM.

Explanation (Optional):

Alarm guidance per 9-5-2/F-3, directs that if drywell pressure cannot be maintained below 1.5 psig, to scram and concurrently enter Procedure 2.1.5. Drywell temperature and pressure is rising which is an indication of a leak in the drywell. With the rate of temperature rise, the pressure in the drywell will be above 1.5 psig very shortly. The key phrase is "if pressure cannot be maintained below 1.5 psig." Cannot be maintained below means action must be taken before the value is reached. Before 1.5 psig is reached procedure guidance requires the reactor to be scrammed.

Distracters:

b. EOP 3A entry on high drywell average temperature is 150°F. Drywell temperature above this level will be reached in approximately 5 minutes at the given rate. There is no EOP 3A entry at this time. The EOPs cannot be entered prematurely, and the condition must be met prior to entry and taking action directed by EOPs. The drywell FCUs are operating and do not trip until drywell pressure rises above 1.84 psig so there is no reason to place them in OVERRIDE. The EOPs and not alarm 9-5-2/F-3 provide guidance for placing the FCUs in OVERRIDE. The candidate that believes the EOPs can be entered early would select this answer. This answer is plausible because EOP 3A does direct the drywell FCU control switches be placed in OVERRIDE in an effort to maintain drywell pressure <1.84 psig.

c. Abnormal Procedure 2.4MC-RF has an entry condition if a leak in the feedwater or condensate system is known. At the rate drywell temperature and pressure are rising, a feedwater leak would be small and the control room operator could not positively determine it to be a feedwater leak. The procedure does direct tripping the Reactor Feedwater pumps if a leak is endangering personnel or safe plant operation. There is no reason to assume a feedwater leak is present with the information given in the question stem. Removing these high pressure pumps from service would not be the correct thing to do at this time. The candidate that assumes a feedwater leak exists would choose this answer but more pressing is the requirement to scram the reactor as stated in 2.4PC.

d. Procedure 2.4PC does direct this action but at a drywell pressure of 0.75 psig. This pressure has long been passed and the more pressing issue is scramming the reactor because of drywell pressure. Lowering RR pump speed to lower power will lower the heat input into the drywell as the RR pumps are a large head load on containment so the operator that is aware of this characteristic would select this answer.

Technical Reference(s):
ARP 2.3_9-3-2, Rev 29
ACP 2.4PC, Rev 17

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT0320128J0J0100, Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Question Source:
Bank # 25631
Modified Bank # (Note changes or attach parent)
New

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

10 CFR Part 55 Content:
55.41 7
55.43

Comments: See original question attached.

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295025.EK2.09	
Importance Rating	3.9	

295025 High Reactor Pressure

EK2. Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:

EK2.09 Reactor Power

Question: 12

The plant is operating at 100% power when an MSIV disc-stem separation occurs on an outboard MSIV.

How does the plant respond in the next 30 seconds?

a. Group 1 isolation.

b. APRM High Flux scram.

c. Low RPV water level scram.

d. SRV opens on spring pressure.

Answer:

b. APRM High Flux scram.

Explanation (Optional):

MSIV stem disc separation results in the MSIV disc rapidly closing causing core steam bubble collapse and a spike in reactor power. This power spike is seen by all the APRMS and a reactor scram on high flux results. The outboard MSIV's are located in the steam tunnel further down the steam lines than the inboard MSIVs so the pressure perturbation is less than an inboard MSIV stem disc separation.

Distracters:

a. is incorrect because the RPS scram signal inserts all control rods. The steam flow through the other steam lines does not appreciably rise due to the scram. The candidate that feels the rise in flow through the other Main Steam Lines may select this answer. This answer is plausible because closing one MSIV at a slower rate will cause the other steam lines to have greater flow.

c. is incorrect because the APRM high flux scram signal is much faster acting than the void collapse. The RPV water level does not lower enough to cause a low level scram signal to be generated. The candidate that does not understand this chain of events may select this answer. This answer is plausible because a high RPV low level inputs to the RPS trip logic.

d. is incorrect because the APRM high flux scram signal shuts the reactor down to decay heat level. The initial pressure spike does not go high enough to open an SRV which is 1080 psig ± 32.4 psig. The candidate that does not understand this chain of events may select this answer. This answer is plausible because rising pressure results from an MSIV stem disc separation.

Technical Reference(s):
COR002-14-02, Rev 23 Main Steam Student Text

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0021402001060D, Given a specific Main Steam system malfunction, determine the effect on any of the following: Reactor power

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level:

Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:

55.41 7
55.43

Comments:

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295026.EA2.02	
Importance Rating	3.8	

295026 Suppression Pool High Temp.

EA2. Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:

EA2.02 Suppression pool level

Question: 13

Following an ATWS the following conditions exist:

The reactor is shutdown.
Reactor pressure is steady at 800 psig.
Average torus water temperature is 200°F and steady.

A torus leak now occurs resulting on a lowering torus water level.

What is the lowest torus level can get before the Heat Capacity Temperature limit curve is exceeded?

a. 13.8 feet

b. 12.9 feet

c. 11 feet

d. 10 feet

Answer:

c. 11 feet

Explanation (Optional):

With RPV pressure constant at 800 psig and average torus water temperature constant at 200°F, the 11 feet line is where the 800 psig curve and the 200°F line intersect. Moving to the right of the 800 psig curve any further is exceeding the Heat Capacity Temperature limit curve.

Distracters:

a. This is where the torus water level; the 1000 psig curve and the 200°F torus temperature line intersect. If the candidate selected the incorrect RPV pressure curve this answer would be chosen. It is reasonable to believe inventory has been added to the torus as a result of the ATWS due to the torus water temperature being so elevated.

b. This is the normal torus water level. This level intersects the 200°F line between the 1000 psig and 800 psig line which is NOT exceeding the Heat Capacity Temperature limit curve. The candidate that decides to use the more conservative 1000 psig curve not knowing that interpolation is allowed with this graph may select this answer. However, the 800 psig curve is the correct one.

d. This level intersects the 600 psig curve beyond the Heat Capacity Temperature limit curve. It intersects with the 800 psig curve at the 190°F torus temperature which is NOT exceeding the Heat Capacity Temperature Limit curve. The candidate that selects the incorrect pressure curve would select this answer.

Technical Reference(s):
INT0080618, Rev 20

Proposed references to be provided to applicants during examination: EOP Graph 7

Learning Objective:
INT00806180010300, Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:
55.41 10
55.43

Comments:

Difficulty: 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295028.EK3.01	
Importance Rating	3.6	

295028 High Drywell Temperature

EK 3. Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE:

EK3.01 Emergency depressurization

Question: 14

What is the reason emergency depressurization is performed when average Drywell Temperature cannot be restored and maintained below 340°F?

a. To ensure the SRVs can still perform this function.

b. To maintain drywell temperature below its design temperature.

c. To ensure the Fuel Zone level instruments remain available for use.

d. To remain within Primary Containment venting capability with all vent valves.

Answer:

a. To ensure the SRVs can still perform this function.

Explanation (Optional):

At CNS, the drywell design temperature is relatively low, 281°F. Replacing the design temperature with a less restrictive value avoids unnecessary emergency depressurization, thereby prolonging RCIC availability.

Emergency RPV depressurization is required in Step DW/T-5 to minimize any direct energy release to the drywell through a primary system break and ensure that SRVs are opened while still operable.

Distracters:

b. Drywell temperature below design is incorrect because this is the reason for spraying the drywell. A candidate may choose this answer if they recall the design temperature of the drywell as being 281°F.

c. Fuel Zone level instruments available for use is incorrect because it's value is 370°F in accordance with the MRT and MIL curves of the EOPs addressed in Caution 1. A candidate may choose this answer if they cannot remember the lower Minimum Run Temperature for the FZ instruments.

d. Primary Containment venting capability with all vent valves is incorrect because this is the reason for venting prior to exceeding Pri. Cont. Pressure Limit A or B. A candidate may choose this answer if they confuse the ADS solenoids with the solenoid operated containment vent valve.

Technical Reference(s):
EOP-3A, EPG Issue 1103

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT00806130011200, Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:
55.41 5
55.43

Difficulty 2

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295030.EK2.09	
Importance Rating	2.5	

295030 Low Suppression Pool Wtr Lvl

EK2.09 Knowledge of the interrelationships between LOW SUPPRESSION POOL WATER LEVEL and the following:

EK2.09 SPDS/ERIS/CRODS/GDS: Plant-Specific

Question: 15

At what level on the Torus Narrow Range Level Instrument does SPDS initially turn Red for Primary Containment when the suppression pool water level starts lowering?

a. -1.0 inches

b. -1.5 inches

c. -2.0 inches

d. -2.5 inches

Answer:

b. -1.5 inches

Explanation (Optional):

SPDS monitors point N356 Narrow Range Torus Water Level and when level drops to -1.5 inches it turns the Primary Containment Alarm Window in SPDS Red to alert the Crew that an EOP entry is imminent.

Distracters:

a. Minus 1.0 inches is incorrect, this is the Alert yellow window setpoint. A candid ate may choose this as the correct answer if they get the yellow and red setpoints confused.

c. Minus 2.0 inches is incorrect, this is the EOP entry setpoint not the red alarm setpoint. A candid ate may choose this as the correct answer if they get the EOP entry setpoint and the red setpoints confused.

d. Minus 2.5 inches is incorrect this is a level that is one inch off of the setpoint but within the acceptable misconception levels. A candid ate may choose this as the correct

answer if they know that the red alarm window setpoint is different by 0.5 inches from the EOP entry setpoint and add it instead of subtracting it.

Technical Reference(s):
Student Text COR0021702, Rev 18

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0021702001110B, Demonstrate the ability to operate a PMIS display console to perform the following activities: Monitor and evaluate plant conditions with Safety Parameter Display System (SPDS) displays.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:
55.41 7
55.43

Comment:

Difficulty 2

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:		
	Level	RO ~~SRO~~
	Tier #	1
	Group #	1
	K/A #	295031 EK1.03
	Importance Rating	3.7

295031 Reactor Low Water Level

EK1: Knowledge of the operational implications of the following concepts as they apply to Reactor Low Water Level:

EK1.03 Water level effects on reactor power

Question: 16

Following a reactor scram several control rods remain in intermediate positions. The following conditions exist:

- Reactor power is 11% and steady.
- Reactor water level is -10 inches and steady.
- Suppression pool temperature is 101°Fand steady.
- Drywell pressure is 1.0 psig and steady.
- Reactor pressure is being controlled by the turbine bypass valves.
- Standby Liquid Control injection is injecting.

Even though NOT all the Level/Power Conditions exist, EOP-7A step FS/L-6 requires reactor vessel water level to be lowered to -60 inches, what is the basis for this step?

Minimize/prevent…

a. boron dilution.

b. torus water heatup.

c. neutron flux oscillations.

d. cold water reactivity addition.

Answer:

c. neutron flux oscillations.

Explanation (Why distractors are incorrect and why correct answer is correct):

During execution of EOP-7A when the level power conditions are not met water level is intentionally lowered to -60 inches to prevent or mitigate the consequences of any large

irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, This level is 24 inches below the elevation of the feedwater sparger nozzles (-60 in.).

a. This option is incorrect because minimizing boron dilution is based on maintaining RPV level below the max normal operating level of +54 inches. Since boron dilution is an operational concern and the reason for some EOP actions a candidate may confuse this with the reason for lowering level to -60 inches and choose this answer.

b. This option is incorrect because no SRVs are open to add heat to the torus and the torus temperature is below the Boron Injection Initiation Temperature (BIIT). But this is the reason for lowering reactor water level when the level power conditions are met so a candidate who does not understand the different reasons for lowering water level may choose this answer as this is the basis for lowering level when level power conditions are present.

d. This option is incorrect because minimizing cold water injection is the major concern when core power is below the heating range and a cold water injection could result in a rapid increase in power before moderator heating could provide negative reactivity feedback. The reactor core is still at power and above the heating range in the condition stated for this question. But because cold water introduction during an ATWS is a major operational concern a candidate may believe that this is the reason for the action.

Technical Reference(s):
Student Text INT008-06-10, Rev 24
EOP Flowchart 7A - RPV Level (Failure to Scram)

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

INT008-06-10
3. State the basis for intentionally lowering RPV water level and the criteria for the lowered level (LL).

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.8 to 41.10
Difficulty:	4

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A #	295037 EK2.12	
Importance Rating	3.6	

295037 Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

EK2: Knowledge of the interrelations between Scram Condition Present and Reactor Power Above APRM Downscale or Unknown and the following:

EK2.12 Rod Control and Information System

Question: 17

An ATWS has occurred with approximately half of the control rods failing to fully insert. Reactor power is 22%. CRD pump 1A is operating. Action is being taken to manually insert control rods.

Which mode of control rod operation is used and why?

a. Continuous In mode in order to bypass the Rod Sequence Timer.

b. Emergency In mode in order to bypass the Rod Sequence Timer.

c. Continuous In mode in order to bypass Rod Worth Minimizer Insert rod blocks.

d. Emergency In mode in order to bypass Rod Worth Minimizer Insert rod blocks.

Answer:

b. Emergency In mode in order to bypass the Rod Sequence Timer.

Explanation (Why distractors are incorrect and why correct answer is correct):

The Emergency In mode is used because it bypasses the Rod Sequence Timer and allows more rapid insertion of the control rods. Bypassing the Rod Sequence Timer eliminates the timed settle function that occurs when a rod motion signal is removed in both the Normal and Continuous In modes of operation and takes several seconds to complete. Selection of another control rod is prevented until the settle function times out. Bypassing this function allows another control rod to be selected and begin inserting immediately.

a. This option is incorrect because Continuous In mode of operation is not used because rod insertion is controlled by the Rod Sequence Timer, which is not bypassed except in the Emergency In mode. The Rod Sequence Timer prevents the selection of another control rod until the rod settle function has timed out allowing the control rod being moved to settle to the next notch position. A candidate may select this option knowing that the rods are to

be driven full in but not knowing that the Rod Sequence Timer is still controlling rod selection and movement, resulting in a slower rod insertion rate.

c. This option is incorrect because Continuous In mode of operation is not used because rod insertion is controlled by the Rod Sequence Timer, which is not bypassed except in the Emergency In mode. Control rod insert blocks from the Rod Worth Minimizer are not automatically bypassed by any control rod operating mode. A candidate may select this option knowing that the rods are to be driven full in and knowing that an insert rod block will prevent rod insertion but not knowing which rod control function is automatically bypassed.

d. This option is incorrect because Emergency In is the correct mode but it does not bypass RWM insert blocks which are bypassed by using a manual keylock function to bypass the RWM. A candidate may choose this option knowing that the rods are to be driven full in and knowing that an insert rod block will prevent rod insertion but not knowing which rod control function is automatically bypassed.

Technical Reference(s):
EOP Flowchart 6A - RPV Pressure & Power (Failure-to-Scram)
Procedure EOP 5.8.3, Rev 16 Alternate Rod Insertion Methods
INT008-06-06, Rev 19 OPS EOP Flowchart 6A - RPV Pressure & Power (Failure-to-Scram)

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

INT008-06-06, OPS EOP Flowchart 6A - RPV Pressure & Power (Failure-to-Scram)
10. Given plant conditions and ESP 5.8.3, ALTERNATE ROD INSERTION METHODS, determine which methods would successfully insert control rods.

Question Source: Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content: 41.6 / 45.8

Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A # 295038	2.1.30	
Importance Rating	4.4	

295038 High Off-Site Release Rate

K/A Contents: 2.1.30 Ability to locate and operate components including local controls

Question: 18

The plant is shutdown with the Turbine Building HVAC shutdown when due to the radioactive release; the declaration of an Alert is required.

The SRO directs the start of the turbine building HVAC as required by EOPs. After the start of two exhaust fans and a supply fan the HV-DPIC-840, TURB BLDG/ATMOS DP, is placed to AUTO.

Turbine Building manual flow control valve HV-FCV-1001, MANUAL FLOW CONTROL VALVE A, is now adjusted.

From what location is this adjustment made and what parameter is being adjusted?

Turbine Building…

a. 932-North; Adjust building flow to 50,000 cfm.

b. 932-North; Adjust building pressure to ≤ -0.25" wg.

c. 903-South; Adjust building flow to 50,000 cfm.

d. 903-South; Adjust building pressure to ≤ -0.25" wg.

Answer:

a. 932-North; Adjust building flow to 50,000 cfm.

Explanation (Why distractors are incorrect and why correct answer is correct):

EOP-5A, Radioactivity Release Control, directs the start of the Turbine Building ventilation system if it is shutdown following entry into EOP-5A. The general sequence of the start of this system is to start two exhaust fans then a supply fan. A that point HV-DPIC-840 is placed to AUTO. The Turbine Building manual flow control valve HV-FCV-1001 is then adjusted from T-932-N (inside Panel LIR-1-HV-T1A) to obtain 50,000 cfm building air flow as read on HV-FR-4000 Channel 3 (VBD-E).

b. This option is incorrect because the valve is adjusted for building flow and not for the differential pressure which is controlled by HV-DPIC-840. However adjusting building pressure is the procedural step following this and therefore a candidate who does not understand how the system functions may believe that the manual flow control valve is adjusted to control DP.

c. This option is incorrect because this is the location of Manual flow control valve HV-FCV-1007 (inside Panel LIR-1-HV-T1C cabinet), But since this is a valve that is manipulated during the start of the system, just not at this point, a candidate may confuse the two locations and choose this answer.

d. This option is incorrect because the valve is adjusted for building flow and not for the differential pressure which is controlled by HV-DPIC-840 and this is the location of Manual flow control valve HV-FCV-1007 (inside Panel LIR-1-HV-T1C cabinet) and not that of HV-FCV-1001.

Technical Reference(s):
COR001-18-01 Rev 24, Radiation Monitoring
Procedure 4.15.1, Rev 23, ELEVATED RELEASE POINT RADIATION MONITORING SYSTEM

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR001-18-01

6. Demonstrate the ability to locate, in the plant, all local indications associated with the HVAC systems and state the significance of each. *

14. Briefly describe the following concepts as they apply to Control Room HVAC:

a. Airborne contamination (e.g., radiological, toxic gas, smoke) control

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.13
Difficulty	3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A # 600000	AK3.04	
Importance Rating	2.8	

600000 Plant Fire on Site

AK3: Knowledge of the reasons for the following responses as they apply to Plant Fire On Site.

AK3.04 Actions contained in the abnormal procedure for plant fire on site.

Question: 19

The plant is operating at power when a fire occurs that eventually requires entry into 5.4FIRE-S/D, FIRE INDUCED SHUTDOWN FROM OUTSIDE CONTROL ROOM.

Before leaving the Control Room 5.4FIRE-S/D requires that the RO/BOP ensure REACTOR MODE switch is in **RUN.**

What is the primary reason the reactor mode switch is required to be in RUN when exiting the Control Room?

a. Ensures the main turbine receives a redundant trip signal.

b. Ensures a second scram signal is provided by MSIV closure.

c. Ensures scram cannot be reset until Control Room is habitable.

d. Ensures scram is delayed to prevent overfilling reactor vessel.

Answer:

b. Ensures a second scram signal is provided by MSIV closure.

Explanation (Why distractors are incorrect and why correct answer is correct):

From procedure 5.4FIRE-S/D,

4.5.2 Before leaving Control Room, RO/BOP perform the following:

<u>NOTE</u> 1 – **REACTOR MODE switch is left in RUN to ensure a secondary scram signal is provided by MSIV closure.**

<u>NOTE</u> 2 – RFLO pumps placed in PULL-TO-LOCK to prevent overfill event from occurring at ~ 50 seconds following scram.

4.5.2.1 SCRAM.

4.5.2.2 Ensure REACTOR MODE switch is in **RUN**.

The mode switch is left in RUN so that in the subsequent steps when the MSIVs are closed a redundant scram signal occurs.

a. This option is incorrect because the main turbine trip is not the reason for leaving the mode switch in RUN. A candidate could confuse the actions of this procedure with those of 5.4ASD which also requires the mode switch be left in RUN before exiting the control room to allow MSIV closure on low pressure which would lead to a turbine trip on anti-motoring.

c. This option is incorrect because although it would prevent reset of the scram this is not the reason for leaving the mode switch in RUN. But a candidate who is unfamiliar with the procedure may reason that with the mode switch in RUN that the scram cannot be reset and therefore believe that this is why it is left in RUN.

d. This option is incorrect because the mode switch position is not likely to affect the reactor water level, however other steps in the procedure are specifically for that reason and a candidate who knows some of the intents of the procedure but not their associations with specific actions may choose this answer.

Technical Reference(s):

Emergency Procedure 5.4 FIRE-SD, Rev 59

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

INT032-01-34, Abnormal Procedures (RO) Fire

D. The trainee will demonstrate the ability to correctly diagnose and assess plant condition(s), and determine the applicable actions required to successfully mitigate a plant event/casualty situation.

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.7

Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	1	
K/A # 700000	AK1.02	
Importance Rating	3.3	

700000 Generator Voltage and Electric Grid Disturbances

AK1: Knowledge of the operational implications of the following concepts as the apply to Generator Voltage and Electric Grid Disturbances:

AK1.02 Over Excitation

Question: 20

The plant is operating at near rated power with the following generator indications:

- Load is at 800 Mw.
- Reactive load is +250 MVAR(out).

Grid oscillations begin and generator excitation rises significantly.

What indication confirms that these grid oscillations are due to a failure in the main generator automatic voltage regulator?

The generator field excitation current rise is coincident with a…

a. rise in indicated 345 kV line voltage only.

b. reduction in indicated 345 kV line voltage only.

c. rise in indicated 345 kV and 161 kV line voltages.

d. reduction in indicated 345 kV and 161 kV line voltages.

Answer:

c. rise in indicated 345 kV and 161 kV line voltages.

Explanation (Why distractors are incorrect and why correct answer is correct):

Per procedure EP 5.3 GRID, oscillations due to a main generator voltage regulator malfunction will typically affect 345 kV and possibly 161 kV grid voltages. Grid generated and voltage regulator failure oscillations could affect 345 kV and 161 kV line voltages. If the grid voltage is oscillating and the voltage regulator is working as designed, as grid voltage lowers, regulator will raise field amps in an attempt to maintain terminal voltage. If the voltage regulator is causing the oscillations, field amps rising will cause grid volts to rise. So, if grid voltage rises as field amps lower, the voltage regulator is working properly. If grid voltage rises as field amps rise,

then the voltage regulator is driving the voltage oscillations. So in this case the cause of the grid voltage oscillation (high) is due to over excitation of the main generator.

a. This option is incorrect because the rise in field curing and the grid is indication the voltage regulator is trying to drive the grid and more than the 345 kV lines are effected. The generator can affect the 345 kV and 161 kV line voltages. This answer is plausible because the 345 kV lines can be effected by a voltage regulator failure.

b. This option is incorrect because, with the field current rising, the grid is falling indicating that the voltage regulator is trying to compensate for the grid voltage fluctuations. A candidate may choose this option as it is a condition indicated in the procedure, particularly if the candidate does not understand the relationship between the grid voltage and field current. This answer is plausible because grid voltage oscillations do effect the voltage regulator response.

d. This option is incorrect because, with the field current rising, the grid is falling indicating that the voltage regulator is trying to compensate for the grid voltage fluctuations. A candidate may choose this option as it is a condition indicated in the procedure, particularly if the candidate does not understand the relationship between the grid voltage and field current. This answer is plausible because grid voltage oscillations do effect the voltage regulator response.

Technical Reference(s):
COR001-13-01 Rev 27, Main Generator and Auxiliaries
Procedure EP 5.3GRID Rev 37

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR001-13-01 R27
5. Given a specific Main Generator and Auxiliaries malfunction, determine the effect on the following:
   a. AC electrical distribution

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.10 / 45.8
Difficulty	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	2	
K/A # 295008	AK2.06	
Importance Rating	3.4	

295008 High Reactor Water Level

AK2: Knowledge of the interrelations between high reactor water level and the following:

AK2.06 RCIC

Question 21

Following a RCIC automatic start, how does the system respond when reactor water level rises above +54 inches?

a. Inboard Steam Line Isolation Valve (MO-15) closes.

b. Outboard Steam Line Isolation Valve (MO-16) closes.

c. The Trip Throttle valve first closes and then reopens once the Steam Supply Block Valve (MO-131) is closed.

d. Steam Supply Block Valve (MO-131) first closes and the Trip Throttle valve then closes due to low oil pressure.

Answer:

d. Steam Supply Block Valve (MO-131) first closes and the Trip Throttle valve then closes due to low oil pressure.

Explanation (Why distractors are incorrect and why correct answer is correct):

In the event of a reactor high water level shutdown, the steam supply block valve (MO-131) closes, then the turbine trip throttle valve closes on low oil pressure caused by the turbine coast down. When both the trip throttle valve and the MO-131 valve are fully closed, the motor-operated trip reset valve automatically resets the trip throttle valve. The RCIC system then automatically restarts on a low water level initiation signal without any operator action required.

a. This option is incorrect because the MO-15 valve closes on a RCIC isolation not high reactor water level. A candidate who confuses RCIC trip automatic actions with those of isolation may choose this answer.

b. This option is incorrect because the MO-16 valve closes on a RCIC isolation, but does not close on RPV high level. A candidate who confuses RCIC isolation and trip signals may choose this answer.

c. This option is incorrect because the Trip Throttle valve does not close first, it closes after MO-131. Since these actions are the same, but in the incorrect sequence. A candidate who knows the actions but not the reason or the sequence may choose this answer.

Technical Reference(s):
Procedure 2.2.67, Rev 68 Reactor Core Isolation Cooling System
COR00218 Rev 22, Reactor Core Isolation Cooling

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR00218R22-S
5. Describe the interrelationship between RCIC system and the following:
   n. Reactor Water Level

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.7 /45.8

Level of Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	2	
K/A # 295009	AK2.01	
Importance Rating	3.9	

295009 Low Reactor Water Level

AK2: Knowledge of the interrelations between Low Reactor Water Level and the following:

AK2.01 Reactor water level indication

Question: 22

Following a plant transient, recirculation pumps have run back to minimum speed and RPV level is 10 inches below the automatic scram setpoint and slowly lowering.

At this time, what water level instrument is appropriate to use to monitor reactor water level?

a. Fuel Zone

b. Wide Range

c. Steam Nozzle

d. Narrow Range

Answer:

b. Wide Range

Explanation (Why distractors are incorrect and why correct answer is correct):

The automatic scram setpoint is +4 inches (≥ +3 inches per Tech Specs) in reference to Instrument Zero (IZ). With water level at 10 inches below the auto scram setpoint and lowering it is at least 6 inches below Instrument Zero.

The Fuel Zone instruments do not read true water level when the recirc or RHR pumps are operating. Since the variable leg of these instruments is piped to the discharge of the jet pumps, the dP cells sense the head of water in the shroud and the head of the jet pumps. The instruments are designed for use during accident conditions, such as a design basis loss of coolant accident, after the recirc pumps have tripped.

Wide Range level instruments in the Control Room use a reference point at 517 in. (Instrument Zero, IZ) and have a range from -155 to +60 in. (IZ).

The Narrow Range level instruments have a range of 0 to 60 in. referenced to IZ.

The Steam Nozzle Range instrument uses the zero reference at IZ and has a range from 0 to +180 in.

a. This option is incorrect. A candidate may select this option if he knows that the Fuel Zone instruments have a range of -320 to +60 inches and does not know (or forgets) the instruments cannot be used if the recirculation pumps are running.

c. This option is incorrect as the instrument cannot read below Instrument Zero. A candidate may select this option if he confuses the automatic scram setpoint with the requirement to manually scram the reactor at +12 inches.

d. This option is incorrect as the instrument cannot read below Instrument Zero. A candidate may select this option if he confuses the automatic scram setpoint with the requirement to manually scram the reactor at +12 inches and knows that the Narrow Range instruments are the most accurate for operating conditions.

Technical Reference(s):
COR0021502 Rev 25, Nuclear Boiler Instrumentation

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

4. Briefly describe the following concepts as they apply to NBI:

   a. Vessel level measurement

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41.7
Difficulty	2

ES-401 **Written Examination Question Worksheet** **Form ES-401-5**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	2	
K/A # 295013	AA1.01	
Importance Rating	3.9	

295013 High suppression pool temp.

AA1: Ability to operate and/or monitor the following as they apply to high suppression pool temperature.

AA1.01: Suppression Pool Cooling

Question: 23

Following a LOCA the following conditions exist:

- RPV water level is steady at -220 inches on the fuel zone instruments.
- Both loops of RHR are in LPCI mode.
- Suppression pool temperature is elevated requiring suppression pool cooling.
- SRO directs the A loop of RHR be placed in Suppression Pool Cooling.

What action(s) is/are **required** in order to allow operation of the A loop Suppression Pool Cooling valves?

a. Place the CONTMT COOLING 2/3 CORE VALVE CONTROL PERMISSIVE switch to MANUAL only.

b. Momentarily place CONTMT COOLING VLV CONTROL PERMISSIVE switch positioned to MANUAL only.

c. Place the CONTMT COOLING 2/3 CORE VALVE CONTROL PERMISSIVE switch to MANUAL and then momentarily place CONTMT COOLING VLV CONTROL PERMISSIVE switch positioned to MANUAL.

d. Momentarily place CONTMT COOLING VLV CONTROL PERMISSIVE switch positioned to MANUAL and then place the CONTMT COOLING 2/3 CORE VALVE CONTROL PERMISSIVE switch to MANUAL.

Answer:

c. Place the CONTMT COOLING 2/3 CORE VALVE CONTROL PERMISSIVE switch to MANUAL and then momentarily place CONTMT COOLING VLV CONTROL PERMISSIVE switch positioned to MANUAL.

Explanation (Why distractors are incorrect and why correct answer is correct):

In order to open the suppression pool cooling valves, Outboard Suppression Pool Cooling Valve MO-39A and Inboard Suppression Pool Cooling Valve MO-34A spray valve control is required

due to the LPCI signal. Since the RPV level is less than 2/3 core height (-193" FZ) the 2/3 core height must first be overridden by then spray valve control obtained by placing the CONTMT COOLING 2/3 CORE VALVE CONTROL PERMISSIVE to MANUAL and then obtaining spray valve control by placing, momentarily, the CONTMT COOLING VLV CONTROL PERMISSIVE switch positioned to MANUAL.

a. This option is incorrect because only overriding the 2/3 core height will not result in spray valve control. The spray valve control switch must also then be taken to manual. A candidate that does not understand spray valve control may choose this answer as they would reason that the suppression pool cooling valves are not spray valves and that only the 2/3 core height need be overridden.

b. This option is incorrect as this action will not result in spray valve control because RPV water level is below 2/3 core height. A candidate who does not realize that -220"FZ is below 2/3 core height would choose this answer because if water lever were above that this would be the correct answer.

d. This option is incorrect because the order of actions is reversed and since the CONTMT COOLING VLV CONTROL PERMISSIVE switch is spring return to normal these actions would not result in spray valve control. Meaning a candidate who does not understand the system may choose this answer knowing the actions but not the required order for the conditions.

Technical Reference(s):
Procedure 2.2.69.3, Rev 46 RHR Suppression Pool Cooling and Containment Spray, Attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0022302R09
17. Given plant conditions, determine actions necessary to place RHR in the following flowpaths:
    a. Torus cooling

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7 /45.6

Level of Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	2	
K/A # 295022	AK1.02	
Importance Rating	3.6	

295022 Loss of CRD Pumps

AK1: Knowledge of the operational implications of the following concepts as they apply to loss of CRD pumps:

AK1.02 Reactivity Control

Question: 24

A reactor startup from cold shutdown is in progress and reactor pressure is 800 psig. The operating CRD pump trips and the alternate pump cannot be started. A restart of the operating pump is attempted but it fails to start.

If these conditions persist, at what point is a manual scram required?

When…

a. the first accumulator low pressure alarm is received.

b. CRD charging header lowers below reactor pressure.

c. the first control rod high temperature alarm is received.

d. it is determined neither CRD pump can be immediately started.

Answer:

d. it is determined neither CRD pump can be immediately started.

Explanation (Why distractors are incorrect and why correct answer is correct):

Per ARP 9-5-2/A-6

1.2 IF both CRD pumps off:

1.2.1 IF Reactor Pressure is < 900 psi with more than one control rod withdrawn, THEN perform following:

1.2.1.1 Attempt immediate start of CRD Pump B.

a. IF CRD Pump B starts, THEN go to Step 1.1.4

1.2.1.2 IF neither CRD pump can be immediately restarted, THEN SCRAM and enter Procedure 2.1.5.

a. This option is incorrect because this condition alone does not require a scram. A candidate may choose this response based on the Tech Spec action allowing 20 minutes to restart a

CRD pump following receipt of the second accumulator trouble alarm.

b. This option is incorrect because this is not the condition that requires a scram, charging water pressure would already be low with the loss of a pump, but if another pump could be started a scram would not be required. A candidate may choose this option if he confuses the slow scram times expected when scramming on reactor pressure alone when reactor pressure is less than 900 psig with the accumulator minimum operability pressure of 940 psig.

c. This option is incorrect. A candidate may choose this option if he confuses the potential for slow scram insertion times with CRD Mechanism temperature above 350 degrees with an inability to insert control rods and knows that high temperatures are an expected condition if cooling water flow cannot be maintained as is the case here.

Technical Reference(s):
COR002-04-02 Rev 23, Control Rod Drive Hydraulics
ARP 9-5-2/A-6, Rev 41 CRD Pump A Breaker Trip

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-04-02

11. Predict the consequences a malfunction of the following would have on the CRDH system:
    i. CRDH pump trip

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.10
Difficulty	4

ES-401 **Written Examination Question Worksheet** **Form ES-401-5**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	2	
K/A # 295033	EK3.01	
Importance Rating	3.3	

295033 High Secondary Containment Area Radiation Levels

EK3: Knowledge of the reasons for the following responses as they apply to secondary containment area radiation levels

EK3.01 Emergency Depressurization

Question: 25

What is a reason that emergency depressurization is required when a primary system is discharging into secondary containment and two secondary containment radiation levels reach their Max Safe Operating Levels?

a. To limit or reduce the offsite radiation release rate.

b. To maintain personnel access to the secondary containment.

c. To prevent damage to equipment necessary for safe shutdown.

d. To maintain access to areas adjacent to secondary containment.

Answer:

b. To maintain personnel access to the secondary containment.

Explanation (Why distractors are incorrect and why correct answer is correct):

Some of the entry conditions for EOP 5A, Secondary Containment Control, are radiation, temperature, or water levels reaching their Max Normal Operating Level (MNO). Emergency Depressurization may be required after entry into the EOP based on parameter escalation to the Max Safe Operating (MSO) levels. In general, the Maximum Safe Operating (MSO) values are defined to be the highest value of a Secondary Containment parameter at which neither equipment necessary for the safe shutdown of the plant will fail nor personnel access necessary for the safe shutdown of the plant will be precluded.
Depressurizing the RPV promptly places the primary system in its lowest possible energy state, rejects heat to the suppression pool in preference to outside the containment, and reduces the driving head and flow of primary systems that are unisolated and discharging into the Secondary Containment. The max safe radiation levels are based on the maintenance of access to secondary containment. The radiation levels, at least in the near term do not affect equipment necessary to safely shutdown the plant.

a. This option is incorrect because the bases for emergency depressurization (ED) when two

areas are at max safe operating level is for access to containment. But since ED is also required in EOP 5A for radiation release a candidate may confuse the bases and choose this answer.

c. This option is incorrect because keeping equipment safe from damage is the reason for emergency depressurization if the secondary containment **temperatures** were at max safe levels.. This is plausible for a candidate who is uncertain about which EOP basis applies for a particular condition.

d. This option is incorrect because inside secondary containment control access to adjacent areas is not the bases for ED. However in EOP 5A radiation release maintenance of access to adjacent areas is addressed in override RR1 so a candidate may confuse these actions and their bases and choose this answer.

Technical Reference(s):
EOP-5A Bases

Proposed references to be provided to applicants during examination: None

Learning Objective:
Student text INT008006 17. Rev 17 EOP 5A
3. Explain why the reactor must be shutdown and depressurized if a secondary containment parameter exceeds its maximum safe operating value in 2 or more areas and the primary system is discharging into secondary containment.

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.5 / 45.6
Level of Difficult:	3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	2	
K/A # 295036	EK2.01	
Importance Rating	3.6	

295036 Secondary Containment High Differential Pressure

EK2: Knowledge of the interrelations between Secondary Containment High Differential Pressure and the following:

EK2.01 Secondary Containment ventilation

Question: 26

The plant is operating at near rated power when alarm REACTOR BLDG HIGH PRESS/ R-2/A-4 alarms. Reactor building pressure is steady at 0 inches of water.

How are reactor building and RRMG ventilation systems affected, if at all?

Reactor Building exhaust, supply, and booster fans…

a. trip immediately; RRMG ventilation remains in operation.

b. and the RRMG ventilation supply and exhaust fans trip immediately.

c. trip after a 45 second time delay; RRMG ventilation remains in operation.

d. and the RRMG ventilation supply and exhaust fans trip after a 45 second time delay.

Answer:

c. trip after a 45 second time delay; RRMG ventilation remains in operation.

Explanation (Why distractors are incorrect and why correct answer is correct):

During normal operations the Reactor Building ventilation system is operated to produce a 0.25 inches of water vacuum (i.e., negative pressure) inside the Reactor Building (compared with the outside atmosphere). Per alarm R-2/A-4, if Reactor Building differential pressure is >-.15" WG or <-.45" WG, the supply, booster, and exhaust fans, if in AUTO, will trip. The RRMG set ventilation system does not trip on high secondary containment pressure, it only trips on a group 6 isolation signal and high differential pressure is not one of those signals, therefore this system continues to operate.

a. This option is incorrect because there is a 45 second time delay before the reactor building fans would trip. A candidate who understands that this high secondary containment pressure trips the reactor building ventilation fans and that who know that RRMG set ventilation continues to operate but who does not recall that the trip has a time delay would

choose this answer.

b. This option is incorrect because there would be a time delay of 45 seconds before the reactor building fans would trip and the RRMG set ventilation would continue to operate. But this answer option is the response that would occur if a group 6 isolation were to occur. It is common misconception to confuse the automatic response of these systems to high differential pressure and a group 6 isolation which is why a candidate may choose this answer.

d. This option is incorrect because even though the Reactor building ventilation system trips after the time delay the RRMG set ventilation system would continue to operate. A candidate who understands that the time delay exists but who incorrectly associates a trip of the RRMG set ventilation system with these conditions would choose this answer.

Technical Reference(s):

COR002-03-02 Rev 28, Containment
COR0010801 Rev 25, Heating, Ventilation and Air Conditioning
ARP 2.3_R-2, Rev 18

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-03-02

7. Describe the interrelationship between Secondary Containment and the following:
   a. Reactor Building Ventilation

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7
Difficulty	3

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A # 295036	2.4.18	
Importance Rating	3.3	

295036 Secondary Containment High Sump/Area Water Level

2.4.18: Knowledge of specific bases for EOPs

Question: 27

The height of what essential equipment provides the bases for the Maximum Safe Operating Level in the torus area?

a. RHR Suction valves

b. Core Spray pump B

c. Suppression Chamber Level Transmitter

d. Torus to Reactor Building Vacuum Breakers

Answer:

c. Suppression Chamber Level Transmitter

Explanation (Why distractors are incorrect and why correct answer is correct):

The maximum safe operating (MSO) torus area water level (4.5 ft, Table 11) corresponds to the height of the suppression chamber level transmitter above the floor. This is the lowest piece of essential equipment in the torus area.

a. This option is incorrect because the valves are normally full open and failure or inaccessibility will not affect RHR operation. A candidate may choose this option knowing that the RHR suction valves being open is critical to protection of the reactor core during an accident and that these valves are situated at a low level in the torus area.

b. This option is incorrect as the B Core Spray pump motor is located above the torus area MSO level. A candidate may choose this option if he knows the drain valve from the Suppression Chamber to the D Sump is procedurally maintained open to assure that SE Quad ECCS leakage does not impair the post-LOCA functionality of Core Spray Pump B and confuses the direction of the drain flow path.

d. This option is incorrect as the vacuum breakers are above the MSO level. A candidate may choose this option if he knows that the torus to reactor building vacuum breakers are critical to primary containment integrity but does not know their elevation in the torus area.

Technical Reference(s):

EOP-5A Bases

INTO080617 Rev 19, EOP Flowchart 5A Secondary Containment and Radioactivity Release Control

COR0020302 Rev 28, Containment

Proposed references to be provided to applicants during examination: None

Learning Objective:

INTO080617 R19

4. State the basis for the limits of the maximum safe operating values (MSO) as they apply to personnel protection and equipment operability.

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.10 /43.1 /45.13

Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401-5**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	1	
Group #	2	
K/A # 203000	K3.03	
Importance Rating	4.2	

203000 RHR/LPCI Injection Mode

K3: Knowledge of the effect that a loss or malfunction of the RHR/LPCI Injection Mode (Plant Specific) will have on following:

K3.03 Automatic Depressurization logic

Question: 28

During an accident the following events occur:

- Reactor water level lowers to ~~<113 inches~~ <— 113 inches EG 8/4/14.
- The only Low Pressure pump that starts is RHR pump A.
- No other injection sources are available.
- ADS actuates and all the ADS valves open.

When reactor pressure falls to 500 psig RHR pump A trips. 15 seconds later RHR pump A is restarted.

How do the ADS valves respond, if at all, to the RHR pump trip and restart?

ADS valves...

a. remain open the entire period.

b. close and remain closed after the pump is restarted.

c. close and reopen 109 seconds after the pump is restarted.

d. close and immediately reopen when the pump is restarted.

Answer:

d. close and immediately reopen when the pump is restarted.

Explanation (Why distractors are incorrect and why correct answer is correct):

The ADS function of the NPR system requires the RHR or CS system to be in operation prior to actuation of the SRVs to reduce reactor pressure. The logic for actuation uses pressure switches in RHR and CS to determine if the system is in operation. Failure of both systems to develop sufficient pressure (setpoint 145 psig) to actuate these switches, thus indicating that the

reactor vessel will not be refilled after depressurization, will prevent ADS automatic initiation or if ADS has actuated then the ADS valves close. Since the ADS timer is timed out when the pump trips the valves close and then immediately reopen when the pump restarts.

a. This option is incorrect because without a low pressure pump available the ADS valves close and remain closed until a Low Pressure pump is operating with discharge pressure. A candidate that believes the ADS logic including the actuation is sealed in once initiated would choose this answer. Since most ECCS logic does seal in to the point of initiating the safety function a candidate may very well believe that this logic is the same and choose this answer. This is a common misconception.

b. This option is incorrect because the valves would reopen when the RHR pump restarts. A candidate who believes that the loss of the pump resets the logic and prevents its re-initiation would choose this answer.

c. This option is incorrect because the ADS valves would reopen immediately with the RHR pump restart. But a candidate may believe that the loss of the pump resets the logic and that the timer would restart and open the valves 109 seconds after the pump restart. This is a plausible misconception as is the answer option A.

Technical Reference(s):
COR002-16-02 Rev 16, Nuclear Pressure Relief
COR002-23-02 Rev 30, Residual Heat Removal System

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR002-16-02
8. Predict the consequences a malfunction of the following would have on the NPR system:
   a. RHR/LPCI system pressure

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7 / 45.4
Difficulty	4

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A #	205000 K5.03	
Importance Rating	2.8	

205000 Shutdown Cooling

K5: Knowledge of the operational implications of the following concepts as they apply to Shutdown Cooling system (RHR Shutdown Cooling Mode):

K5.03 Heat Removal Mechanisms

Question: 29

The reactor is shutdown and B loop of RHR is in shutdown cooling mode of operation. Coolant temperature is currently 275°F and the cooldown rate is currently **95°F/hr.**

With the cooldown rate at 95°F/hr, what action is appropriate to adjust the cooldown rate as required?

d. Lower coolant flow rate through the RHR heat exchanger.

b. Raise the coolant flow rate through the RHR heat exchanger.

a. Lower Service Water flow rate through the RHR heat exchanger.

c. Raise the Service Water flow rate through the RHR heat exchanger.

Answer:

a. Lower coolant flow rate through the RHR heat exchanger.

Explanation (Why distractors are incorrect and why correct answer is correct):

Per procedure 2.2.69.2, the RPV heatup or cooldown rate is controlled at less than or equal to 90 degrees F/hr by adjusting the coolant flow rate through the RHR heat exchanger. More coolant flow through the heat exchanger raises the cooldown rate. Less flow lowers the cooldown rate. The Service Water side flow rate is adjusted to maintain heat exchanger shell to tube differential pressure and possible partition sheet buckling.

b. This option is incorrect because this would raise the cooldown rate and the rate needs lowering. If the student only remembered the TS cooldown rate 100 degrees F/hr limitations he would choose this answer. This answer is plausible because this method of changing the cooldown is is utilized, just not in this instance.

c. This option is incorrect because this action would raise the tube to shell dP and there is no information stating this action is required. Lowering the cooling medium would lower the cooldown rate but this is not the method directed by the procedure.

d. This option is incorrect because this action would lower the tube to shell dP and there is no information stating this action is required. Raising the cooling medium would raise the cooldown rate which is the wrong thing to do. This is not the method directed by the procedure for adjusting cooldown rate anyway.

Technical Reference(s):
COR002-23-02 Rev 30, Residual Heat Removal System
COR002-27-02 Rev 36, Service Water System
Procedure 2.2.69.2, Rev 86 RHR System Shutdown Operations

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-23-02
6. Given an RHR control manipulation, predict and explain changes in the following:
   a. Heat exchanger temperature and flow

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.5 / 45.3
Difficulty	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 206000	A1.08	
Importance Rating	4.1	

206000 HPCI

A1: Ability to predict and/or monitor changes in parameters associated with operating the HPCI controls including:

A1.08 System line-up

Question: 30

The plant is operating with the following conditions:

- Reactor power is 100%.
- Reactor Pressure is 1010 psig and steady.
- Main Steam equalizing header pressure is 955 psig and steady.
- High Pressure Coolant Injection system (HPCI) is isolated due to an instrument fault that caused a PCIS Group 4 isolation.
- HPCI Turbine Inlet Steam Pressure is 0 psig.

The instrument fault is repaired and HPCI is being restored to a normal standby lineup in accordance with 2.2.33 HIGH PRESSURE COOLANT INJECTION SYSTEM operating procedure.

1) What order are the isolation valves opened?
2) To what steam pressure does the HPCI steam line stabilize?

a. 1) HPCI-MO-15 (Inboard) is opened and then MO-16 (Outboard).
   2) Steam pressure stabilizes at approximately 955 psig.

b. 1) HPCI-MO-16 (Outboard) is opened and then MO-15 (Inboard).
   2) Steam pressure stabilizes at approximately 955 psig.

c. 1) HPCI-MO-15 (Inboard) is opened and then MO-16 (Outboard).
   2) Steam pressure stabilizes at approximately 1010 psig.

d. 1) HPCI-MO-16 (Outboard) is opened and then MO-15 (Inboard).
   2) Steam pressure stabilizes at approximately 1010 psig.

Answer:

d. 1) HPCI-MO-16 (Outboard) is opened and then MO-15 (Inboard).
   2) Steam pressure stabilizes at approximately 1010 psig.

Explanation (Why distractors are incorrect and why correct answer is correct):

From procedure 2.2.33, HPCI:

6.6 WHEN reactor pressure is ≥ 107 psig, THEN warm turbine and steam supply lines by placing the HPCI steam supply line in service as follows:

6.6.1 Ensure HPCI-AO-42 and HPCI-AO-43, STM LINE DRAIN TO CNDSR ISOL VLVs, are open.

6.6.2 Ensure MS-TP-17, RHR HX A STM SUPPLY DRIPLEG DRAIN TRAP, is valved into service.

6.6.3 Ensure HPCI-MO-15, STM SUPP INBD ISOL VLV, is closed.

6.6.4 Open HPCI-MO-16, STM SUPP OUTBD ISOL VLV.

6.6.5 Slowly (jog) open HPCI-MO-15 and warm steam supply line.

As HPCI-MO-15 is opened the HPCI steam line pressurizes up to approximately reactor pressure because the HPCI steam line taps off the steam header upstream of the MSIVs and since there is no flow in the HPCI system following pressurization there would be no head loss in the line so HPCI turbine inlet pressure would rise to approximately reactor pressure.

a. This option is incorrect because pressure would rise to reactor pressure and the valve opening sequence is incorrect. The candidate who does not understand where the HPCI steam supply is in relation to the MSIVs may choose this answer as this is the pressure of the equalizing header that is downstream of the actual HPCI tap. This answer is plausible because one valve is opened at a time.

b. This option is incorrect because pressure would rise to reactor pressure. The candidate who does not understand where the HPCI steam supply is in relation to the MSIVs may choose this answer as this is the pressure of the equalizing header that is downstream of the actual HPCI tap. This answer is plausible because the correct valve sequence is listed.

c. This option is incorrect because the outboard valve is opened first. The candidate who believes inboard valve is opened first would choose this answer. This answer is plausible because the correct steam line pressure is listed.

Technical Reference(s):

Procedure 2.2.33, Rev 74 HIGH PRESSURE COOLANT INJECTION SYSTEM
Procedure 2.1.1, Rev 171 Startup Procedure

Proposed references to be provided to applicants during examination: None

Learning Objective: 10b COR0021102R29

COR0021102R29

5. Describe the interrelationship between HPCI and the following:

c. Reactor pressure

SKL012-43-01

7. o. Place the HPCI system in standby status IAW SOP 2.2.33.

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41.7 / 45.5
Difficulty	3

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A # 209001	A3.02	
Importance Rating	3.8	

209001 LPCS

A3: Ability to monitor automatic operations of the LPCS including:

A3.02 Pump starts

Question: 31

The plant is operating when a loss of coolant accident (LOCA) occurs. Plant busses transfer as designed and offsite power remains available.

A few seconds after the LOCA begins drywell pressure rises above 1.84 psig and 1 minute later reactor water level lowers below -113 inches.

At what point do the Core Spray pumps automatically start?

a. Immediately when drywell pressure rises above 1.84 psig.

b. 10 seconds after drywell pressure rises above 1.84 psig.

c. Immediately when reactor water level lowers to -113 inches.

d. 10 Seconds after reactor water level lowers to -113 inches.

Answer:

b. 10 seconds after drywell pressure rises above 1.84 psig.

Explanation (Why distractors are incorrect and why correct answer is correct):

CS pumps automatically start after a 10 second time delay on high drywell pressure <u>or</u> low low low Reactor water level with critical bus power available. Since the critical buses remain available in this case the core spray pumps start 10 seconds after the drywell pressure rises above 1.84 psig. The 1.84 psig is a Technical Specification limit and the student answers questions based on TS numbers.

a. This option is incorrect because the pumps do not start immediately at 1.84 psi there is a 10 second delay. But a candidate may believe that since power remained on the busses that no delay would occur and would therefore choose this answer.

c. This option is incorrect because the pump would already be running by the time the water level lowered to this point. A candidate who does not recall that drywell pressure in addition to low low low water level and that the pumps start following a 10 second delay would

choose this answer.

d. This option is incorrect the CS pumps would already be operating. A candidate who recalls that there is a 10 second delay but who does not know that DW pressure also starts the pump would choose this answer.

Technical Reference(s):
Procedure 2.2.9, Rev 76 CORE SPRAY SYSTEM
ARP 9-5-2/F-3, Rev 41 High Drywell Pressure

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR020602R21
12. Given plant conditions, determine if any of the following Core Spray actions should occur:
    b. Pump starts

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.7 / 45.7

Difficulty 2

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	_____
Group #	1	_____
K/A # 211000	K2.02	
Importance Rating	3.1	

211000 SLC

K.2 Knowledge of electrical power supplies to the following:

K2.02: Explosive Valves

Question 32

What is the power supply that is used to fire the "B" Standby Liquid Control (SLC) squib valve?

a. MCC K

b. MCC S

c. MCC M

d. 120 VAC CPP

Answer:

b. MCC S

Explanation (Why distractors are incorrect and why correct answer is correct):

MCC S is the power supply to the B SLC pump and the squib valve receives its power from the pump supply breaker.

Each of the possible answers is a listed power supply for SLC components and is therefore plausible if the candidate is uncertain of the correct answer.

a. This option is incorrect because this is the power supply for SLC squib valve A and may be selected if the candidate is confused as to which source supplies which valve.

c. This option is incorrect because this is the listed power supply for SLC heat tracing. A candidate that knows SLC loads are powered from MCC M but is not certain of the squib valve power supply may choose this answer because of its association with the SLC system.

d. This option is incorrect because this is the power supply for the squib valve ready lights. A candidate that does not know the squib valves are fired by an auxiliary contact in the pump breaker may believe this circuit also powers the squib valves since it does provide power to a squib related component (squib ready lights). **Note: that even though this power supply appears different than the other options it is highly plausible because it powers actual squib components just not the power to fire the squib.**

Technical Reference(s):

Ops Standby Liquid Control/COR0022902 Rev 20
Procedure 2.2.23, Rev 53 120/240 VAC Instrument Power Distribution System

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022902 Rev 20
13. State the electrical power supply to the following SLC components:
    b. Squib valves

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.7
Difficulty:	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	_____
Group #	1	_____
K/A # 211000	A2.05	
Importance Rating	3.1	

211000 SLC

A.2 Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.05 Loss of SBLC tank heaters

Question 33

The supply breaker for the SLC tank heater is tripped and must be repaired. What is the potential impact on the SLC system and if this condition persists; what corrective action is required?

a. Boron precipitating out of solution; start a SLC pump and recirculate storage tank.

b. Boron precipitating out of solution; raise SLC area ambient temperature with heaters.

c. Boron crystalizing in SLC suction piping; start a SLC pump and recirculate storage tank.

d. Boron crystalizing in SLC suction piping; raise SLC area ambient temperature with heaters.

Answer:

b. Boron precipitating out of solution; raise SLC area ambient temperature with heaters.

Explanation (Why distractors are incorrect and why correct answer is correct):

ARP 9-5-2/G-8 is the alarm response procedure for loss of the SLC tank heater, and directs corrective action to be taken per the low temperature section of Procedure 2.2.74. The required action is to power area heaters from 480 VAC receptacles to raise ambient temperature.

a. This option is incorrect because the tank is not recirculated for a loss of a heater. But a candidate may believe that recirculating the water would mix the tank contents and keep the boron in solution.

c. This option is incorrect because boron crystallization in the suction piping would be the impact of losing power to the heat tracing, not the tank heaters and because the tank is not recirculated for a loss of the tank heater. But a candidate may believe that it is the

suction piping that is impacted and that recirculating the water would mix the tank contents and keep the boron in solution.

d. This option is incorrect because boron crystallization in the suction piping would be the impact of losing power to the heat tracing not the tank heater but a candidate could believe that boron crystallization in the suction pipe would be the primary concern.

Technical Reference(s):

Ops Standby Liquid Control/COR0022902 Rev 20
AR 9-5-2/G-8, Rev 41
Procedure 2.2.74 Rev 48, Standby Liquid Control System

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022902 R20

10. Predict the consequences a malfunction of the following would have on the SLC system:
    g. Tank Heaters

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41.10/ 45.6
Difficulty:	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 212000	K1.06	
Importance Rating	3.5	

212000 RPS

K1: Knowledge of physical connections and/or cause-effect relationship between RPS and the following:

K1.06 CRDH

Question: 34

When an RPS scram signal occurs, do the scram pilot solenoid valves and the backup scram solenoid valves energize or de-energize?

	Scram Pilot Solenoid Valves	**Backup Scram Solenoid Valves**
a.	energize	energize
b.	de-energize	de-energize
c.	de-energize	energize
d.	energize	de-energize

Answer:

c.	de-energize	energize

Explanation:

A dual solenoid scram pilot valve, located on the Control Rod Drive Hydraulic (CRDH) system Hydraulic Control Unit (HCU) for each control rod, supplies air to the scram inlet and outlet valves for each control rod. One scram pilot solenoid is powered by RPS trip system A, the other scram pilot solenoid is powered by RPS trip system B. With either solenoid energized, air pressure holds the scram valves closed. If both solenoids de-energize, air is vented from the scram air header and the scram valves will open, inserting the control rod to shutdown the reactor.

Two DC solenoid operated backup scram valves energize to open when a scram signal is received. Opening of the backup scram valves vents the entire scram air header and allows the scram inlet and outlet valves for all control rods to open, inserting all of the control rods. The backup scram valves are energized (initiate scram) when both trip system A and trip system B are tripped.

Distractors are plausible if the candidate does not understand that the RPS trip function de-energizes the scram pilot solenoids and a trip relay which closes a B contact to energize the backup scram valves from DC power.

a. This option is incorrect because, although the backup scram valves do in fact energize, the scram pilot valves de-energize. A candidate that does not understand that this system is designed to actuate when power is lost may choose this answer.

b. This option is incorrect because even though the scram pilot valves de-energize the back up scram valves energize from DC power when an RPS actuation occurs. This option is highly plausible because in systems like RPS the components are designed to de-energize upon actuation but the backup scram valves energize.

d. This option is incorrect because these are exactly backward in that the scram solenoid pilot valves de-energize and the back up scram valves energize. A candidate that confuses the two different actuation conditions would choose this option.

Technical Reference(s):
Procedure 4.5, Rev 31 RPS Instrument Operating Procedure

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022102 Rev 23, Reactor Protection System
10. Describe the interrelationship between the RPS and the following:
g. CRDH

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.6 / 45.7 to 45.8

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 212000	K4.12	
Importance Rating	3.9	

212000 RPS

K4: Knowledge of RPS design features and/or interlocks which provide for the following:

K4.12 Bypassing of selected Scram signals (manually and automatically)

Question: 35

With the plant operating at low power, the Reactor Mode Switch is taken from RUN to STARTUP.

What scram signal is automatically bypassed by this action?

a. MSIV Closure

b. Reactor High Pressure

c. Turbine Control Valve Fast Closure

d. Scram Discharge Volume Water High Level

Answer:

a. MSIV Closure

Explanation (Why distractors are incorrect and why correct answer is correct):

The MSIV closure scram is bypassed any time the Reactor Mode Switch is in any position other than "RUN". This bypass allows for all four steam lines to be isolated and the reactor to be operated in a hot standby condition at low power levels. This condition occurs during reactor startups and certain reactivity tests during refueling.

b. This option is incorrect because reactor high pressure scram is not bypassed by taking the mode switch out of RUN but a candidate may believe that at low power the high pressure scram is not needed because overpressure protection is provided by high APRM fixed scram signal at 14.5% reactor power.

c. This option is incorrect because the Turbine Control Valve Fast Closure scram signal is bypassed when reactor power is less than 29.5% as sensed by turbine 1st stage pressure. A candidate may choose this option if he is not sure of the correct answer but knows the Turbine Stop Valve and Turbine Control Valve Fast Closure signals are not required at low power and remembers that reactor power cannot exceed 14.5% with the Mode switch not in Run.

d. This option is incorrect. The Scram Discharge Volume Water High Level scram signal is bypassed only when the reactor mode switch is in Shutdown or Refuel <u>and</u> the keylock switch is in Bypass. A candidate may choose this option if he is not sure of the correct answer but remembers this scram signal cannot be bypassed with the reactor mode switch in the Run position.

Technical Reference(s):
COR0022102 Rev 21, Reactor Protection system

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0022102
4. Describe the RPS design features and/or interlocks that provide for the following:
   j. Bypassing of selected scram signal (manually and automatically)

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.7
Difficulty	2

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	_____
Group #	1	_____
K/A # 215003	A4.06	
Importance Rating	3.0	

215003 IRM

A.4 Ability to manually operate and/or monitor in the control room:

A4.06 Detector drives

Question 36

A plant shutdown is in progress and generator output is at 100 MWe. All 8 Intermediate Range Monitor (IRM) detectors are selected and the "Drive-in" pushbutton depressed.

What is an action or condition that stops IRM detector insertion?

a. IRM RPS Upscale Trip.

b. Taking the Reactor Mode Switch out of RUN.

c. Momentarily depressing the "Drive In" pushbutton.

d. Momentarily depressing the "Drive Out" pushbutton.

Answer:

d. Momentarily depressing the "Drive Out" pushbutton.

Explanation (Why distractors are incorrect and why correct answer is correct):

The DRIVE IN push button is a momentary contact type switch with a latching contact around it. Once pushed, the contact closes and the detector moves in regardless of the DRIVE IN push button position. The detector will stop when the FULL IN limit switch is actuated. To stop detector insertion, momentarily press the DRIVE OUT button. This will de-energize the relays associated with the DRIVE IN control circuit. The DRIVE IN signal may be removed at any time by momentarily depressing the DRIVE OUT push button or by depressing the POWER ON push button to de-energize the detector drive circuit. The DRIVE IN signal is also removed if power to the circuit from CPP-2 is lost.

a. This option is incorrect because an IRM upscale trip will initiate a rod block and reactor scram signal but will not affect detector drive motion. Since the IRM upscale is a significant operational concern, a candidate who does not understand the drive in circuit or the actions initiated by an IRM upscale may select this option.

b. This option is incorrect because taking the mode switch out of RUN will initiate a rod withdrawal block if any IRM detector is not fully inserted, but will not stop detector motion. Placing the mode switch in RUN automatically bypasses IRM rod block and trip signals if reactor power is above the APRM downscale setpoint. A candidate may select this option if confused on the interlock and trip functions between the IRMs and reactor mode switch position. Particularly because the mode switch is interlocked with detector position.

c. This option is incorrect because the "Drive-in" pushbutton has a latching contact around it that provides a seal-in function for inserting detectors. Depressing the button again will not stop drive motion. A candidate who believes that this switch is unlatched by re-depressing it would select this option.

Technical Reference(s):

COR0021202 Rev 13, Intermediate Range Monitor
COR0023002 Rev 13, Source Range Monitor
Procedure 4.1.2 Rev 21, Intermediate Range Monitoring System

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0021202 R13

5. Describe the IRM system design features and/or interlocks that provide the following:

   e. Changing detector position

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.6 / 45.5 to 45.8

Difficulty 2

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A #	215004 A4.01	
Importance Rating	3.9	

215004 Source Range Monitor

A.4 Ability to manually operate and/or monitor in the control room:

A4.01 SRM count rate and period

Question 37

Control rods are being withdrawn to take the reactor critical during a plant startup. Alarm 9-5-1/F-8, SRM Period is received for Source Range Monitor (SRM) D <u>only</u>. All SRM period meters indicate the period is becoming shorter.

Which action is required?

a. Bypass SRM D.

b. Insert control rods.

c. Manually scram the reactor.

d. Withdraw the detector for SRM D.

Answer:

b. Insert control rods.

Explanation (Why distractors are incorrect and why correct answer is correct):

Per alarm procedure 9-5-1/F-8, SRM Period, IF an SRM Period alarm is received and the period is getting shorter THEN control rods are inserted to turn the reactor period.

a. This options is incorrect because the required action is to insert control rods. Although a single SRM is high it may indicate a local criticality. A candidate may choose this option if he believes that having only SRM D indicating a short period means the channel is faulty.

c. This option is incorrect because a reactor scram is not required. A candidate may choose this option if he understands a short period alarm indicates a potential rapid power increase but does not know the correct action.

d. This option is incorrect because this is an appropriate action when the reactor power is ascending and SRM indication is high. But in this case the reactor is approaching critical

so the SRM must remain fully inserted. A candidate who recalls that SRMs are withdrawn when they indicate high levels after criticality is reached and power is raised may choose this option.

Technical Reference(s):

Source Range Monitor/COR0023002, Rev 13
GOP 2.1.1, Rev 171 Startup Procedure.
ARP 9-5-1/G-8, Rev 27SRM Retract Not Permitted

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0023002 Rev 13

5. Given an SRM control manipulation, predict changes in the following:

   e. Period alarm trip setpoints

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.6 / 45.5 to 45.8

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 215005	A1.03	
Importance Rating	3.6	

215005 APRM / LPRM

A1: Ability to predict and/or monitor changes in parameters associated with operating the APRM/LPRM controls including:

A1.03 Control rod block status.

Question: 38

The plant is operating at near full power when the Function Switch for APRM Flow Unit A is taken out of OPERATE.

What automatic action(s) occur(s) as a result?

a. Rod block only.

b. Half Scram only.

c. A rod block and a half scram only.

d. APRM INOP, rod block and half scram.

Answer:

a. Rod block only.

Explanation (Why distractors are incorrect and why correct answer is correct):

If a flow unit's Function Switch is taken out of OPERATE in either flow unit, its upscale trip relay trips. This causes a rod block, a "FLOW REF OFF NORMAL" annunciator, an "UPSC or INOP" white indicating light on Panel 9-5, an "INOP" white indicating light on the respective flow unit, and "UPSC/INOP" amber indicating light on Panel 9-14. This rod block is generated by the RPS A channel APRMs.

b. This option is incorrect because no half scram occurs. But for other APRM and Neutron Monitoring systems, it is normal for a half scram to be generated when the associated function switch is taken out of operate which is why a candidate may choose this option.

c. This option is incorrect because even though a rod block occurs, no half scram occurs. But for other APRM and Neutron Monitoring systems, it is normal for a half scram (and usually a rod block) to be generated when the associated function switch is taken out of operate which is why a candidate may choose this option.

d. This option is incorrect because this does not generate an APRM INOP. This distractor

is highly plausible because first an APRM inop does generate a rod block and a half scram and secondly there are a number of items that cause the APRM to generate an INOP. Including loss of inputs and the APRM function switch out of operate. Since both of these concepts are in the test item a candidate may choose this answer.

Technical Reference(s):

Procedure 4.1.3, Rev 25 AVERAGE POWER RANGE MONITORING SYSTEM
COR0020102, Rev 24 APRM System

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0020102, Rev 24
8. Describe the APRM design feature(s) and/or interlock(s) that provide for the following:
   a. Rod withdrawal blocks

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.5 / 45.5

Level of Difficulty 2

ES-401 **Written Examination Question Worksheet** **Form ES-401-5**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 215005	2.2.44	
Importance Rating	4.2	

215005 APRM / LPRM

2.2.44: 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.

Question: 39

The plant is operating steady state at full power with APRM F bypassed, when Annunciator 9-5-1/B-8, APRM UPSCALE is received and does not clear. Investigation shows APRM A reading 112% power with APRMs B, C, D, and E reading 100%.

What automatic action(s) occur(s) and what action is appropriate?

Rod withdrawal block…

a. only; bypass APRM A.

b. only; reduce reactor power.

c. and 1/2 scram on RPS channel A; bypass APRM A.

d. and 1/2 scram on RPS channel A; reduce reactor power.

Answer:

a. only; bypass APRM A.

Explanation (Why distractors are incorrect and why correct answer is correct):

Per ARP 9-5-1/B-8

1. AUTOMATIC ACTIONS

   Rod block.

   OPERATOR OBSERVATION AND ACTION

   Determine from remote indicators and recorders on Panel 9-5 which APRM(s) is affected.

   IF only one APRM per RPS channel is high, THEN bypass that APRM.

   Reduce power if all APRMs are high.

b. This option is incorrect as reducing reactor power is not necessary. A candidate may choose this option if he does not know the correct action to take for a single APRM reading upscale or if confused by APRM F being in bypass.

c. This option is incorrect as a 1/2 scram will not occur. A candidate may choose this option if he doesn't know the APRM trip setpoints and thinks a 1/2 scram should occur at 112% power.

d. This option is incorrect as a 1/2 scram will not occur. A candidate may choose this option if he doesn't know the APRM trip setpoints and thinks a 1/2 scram should occur at 112% power, or if he does not know the correct action to take for a single APRM reading upscale.

Technical Reference(s):
Procedure 4.1.3, Rev 25 AVERAGE POWER RANGE MONITORING SYSTEM
COR0020102 Rev 24, Average Power Range Monitor
ARP 9-5-1/B-8, Rev 27 APRM UPSCALE

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0020102, Rev 24
8. Describe the APRM design feature(s) and/or interlock(s) that provide for the following:
   a. Rod withdrawal blocks

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.6/43.5 / 43.6 / 45.1

Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 217000	A2.16	
Importance Rating	3.5	

217000 RCIC

A2: Ability to (a) predict the impacts of the following on the Reactor Core Isolation Cooling system (RCIC) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.16 Low condensate storage tank level.

Question: 40

RCIC is operating providing makeup flow to the reactor vessel. Level in Emergency Condensate Storage Tank (ECST) "A" is at 26 inches and lowering.

As the level in the ECST continues to lower, how does RCIC respond, and what action is required next?

41 RG 3/4/14

RCIC suction valve MO-~~14~~ will <u>only auto realign</u> to the suppression pool at 24 inches...

a. in BOTH ECST tanks; provide makeup to the ECST tanks.

b. in EITHER ECST tank; manually close the system test return valves (MO-30 and MO-33).

c. in BOTH ECST tanks; manually close the system test return valves (MO-30 and MO-33).

d. in EITHER ECST tank; provide makeup to the ECST tanks.

Answer:

d. in EITHER ECST tank; provide makeup to the ECST tanks.

Explanation (Why distractors are incorrect and why correct answer is correct):

RCIC suction is aligned to a common suction header from the ECSTs in a normal operating or standby lineup. The ECSTs are cross-connected and will therefore be at the same level.

Suppression Pool Suction Valve (MO-41) is a normally closed, motor-operated valve, powered from the 125V DC RCIC starter rack. This valve will automatically open when the water level in <u>either ECST</u> decreases to 24" above the tank's bottom (≥23" per Tech Specs).

When the Suppression Pool suction valve opens, the full open limit switches cause the ECST suction valve (MO-18) and the RCIC test return valves (MO-30 and 33) to close. Per ARP 9-4-1/F-2, when the suction transfer alarm is received, action should be taken to provide makeup to the ECST per procedure 2.2.7.

a. This option is incorrect because only one ECST tank being below 24 inches will realign the suction path. A candidate who believes that both the ECSTs must be low may choose this answer. This answer is plausible because the correct action to provide makeup to the tanks is listed.

b This option is incorrect because the system test return valves automatically close as the torus suction valve opens. There is no requirement to close these valves manually. The candidate that does not recall the valve interlocks may select this answer. This answer is plausible because the correct system response is listed.

c This option is incorrect because only one ECST tank being below 24 inches will realign the suction path. A candidate who believes that both the ECSTs must be low may choose this answer. The test return valves automatically close as the torus suction valve opens. There is no requirement to close these valves manually. The candidate that does not recall the valve interlocks may select this answer. This answer is plausible because test return valves do close on a low ECST tank level even though it is automatically.

Technical Reference(s):
COR002-18-02 Rev 22, Reactor Core Isolation Cooling
Procedure 2.2.67.1, Rev 31 Reactor Core Isolation Cooling System Operations
ARP 9-4-1/F-2, Rev 46 RCIC Suction Transfer

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-18-02
12. Given plant conditions, determine if the following RCIC actions should occur:
    c. ECST suction transfer

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7
Difficulty	3

ES-401 **Written Examination Question Worksheet** **Form ES-401-5**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 218000	A3.07	
Importance Rating	3.7	

218000 ADS

A3: Ability to monitor automatic operation of the ADS including:

A3.07 ADS lights and alarms.

Question: 41

The plant is operating when an accident occurs that causes reactor water level to drop. Initially no Core Spray or RHR pumps can be started. The following events occur:

- Reactor water level falls below -113 inches.
- One minute later an RHR pump is started.

At what point does alarm ADS TIMERS ACTUATED, 9-3-1/A-1 annunciate; and when ADS actuates, what ADS valve indications are illuminated on panel 9-3?

The alarm annunciates when…

a. the RHR pump starts; only the Amber light is illuminated.

b. the RHR pump starts; the Amber and Red lights are illuminated.

c. water level is below -113 inches; only the Amber light is illuminated.

d water level is below -113 inches; the Amber and Red lights are illuminated.

Answer:

d water level is below -113 inches; the Amber and Red lights are illuminated.

Explanation (Why distractors are incorrect and why correct answer is correct):

When water level drops below -113 inches the ADS Timer starts and if a low pressure system is operating when the timer times out (109 seconds) then ADS Actuates. If no LP system is operating when the timer times out then ADS does not actuate.

When ADS actuates it opens the SRVs by energizing their respective open solenoids. When the solenoid energizes the red light associated with the valve illuminates. When the SRV opens the pressure in the tail pipe rises and closes a pressure switch that energizes the Amber light associated with the SRV providing positive indication of valve opening. So for ADS actuation both the red light and the amber light are illuminated.

a. This option is incorrect because the red light associated with the valve is also illuminated because ADS energizes the open solenoid for the valve. Additionally the timer actuated alarm energizes when the level falls below -113 inches not when the RHR pump started. But since both the low level and the low pressure pump are required to actuate ADS a candidate may very well confuse this with the logic to actually actuate the timers. Additionally this candidate believes that only the Amber light is illuminated when ADS actuates which is reasonable to believe as when the SRVs lift due to high pressure only the amber light illuminates.

b. This option is incorrect because the timer actuated alarm energizes when the level falls below -113 inches not when the RHR pump started. But since both the low level and the low pressure pump are required to actuate ADS a candidate may very well confuse this with the logic to actually actuate the timers.

c. This option is incorrect because the red light associated with the valve is also illuminated because ADS energizes the open solenoid for the valve. This candidate believes that only the Amber light is illuminated when ADS actuates which is reasonable to believe as when the SRVs lift due to high pressure only the amber light illuminates.

Technical Reference(s):
Procedure 2.4SRV, Rev 15 STUCK OPEN RELIEF VALVE
ARP 9-3-1/A-2 , Rev 31 Relief Valve Open
COR0021602 Rev 16, Nuclear Pressure Relief

Proposed references to be provided to applicants during examination: None

Learning Objective: 7.o
COR0021602
6. Briefly describe the following concepts as they apply to NPR:
   e. Safety/Relief Valve tailpipe temperature/pressure relationship

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7 / 45.7
Difficulty	3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 223002	K6.06	
Importance Rating	2.8	

223002 PCIS/Nuclear Steam Shutoff

K6: Knowledge of the effect that a loss or malfunction of the following will have on the Primary Containment Isolation system/Nuclear Steam Supply Shutoff:

K6.06 Various process instrumentation

Question: 42

The plant is operating at 100% power and the Reactor Core Isolation Cooling (RCIC) system is in a normal standby lineup. RCIC-DPIS-84, RCIC STEAM LINE HIGH FLOW CHANNEL B, fails and is providing maximum steam line flow input to logic channel B.

How do RCIC valve(s) respond to this failure?

a. Only RCIC-MO-15, Inboard Isolation Valve, closes.

b. Only RCIC-MO-16, Outboard Steam Supply Isolation Valve, closes.

c. RCIC-MO-15, Inboard Isolation Valve, closes, and RCIC-MO-27, Minimum Flow Valve, opens.

d. RCIC-MO-16, Outboard Isolation Valve, and RCIC-MO-27, Minimum Flow Valve, close.

Answer:

a. Only RCIC-MO-15, Inboard Isolation Valve closes.
.

Explanation (Why distractors are incorrect and why correct answer is correct):

RCIC-DPIS-84 provides steam flow input to the B isolation logic for RCIC. When the instrument fails upscale the B logic is actuated giving a half group 5 isolation. The Minimum Flow Valve MO-27 closes on a system isolation if the system is in operation but that comes from the A logic. The minimum flow valve is closed in standby lineup.

Per procedure 2.1.22,

8.1 Upon 1/2 Group 5 Isolation, following will occur:

NOTE – Manual actuation of Group 5 Isolation (Panel 9-4) trips Logic A only.

8.1.1 If Logic A trips:

8.1.1.1 RCIC-MO-16, OUTBD STM SUPP ISOL VLV, closes.

8.1.1.2 RCIC turbine trips.

8.1.2 If Logic B trips, RCIC-MO-15, INBD ISOL VLV, closes.

b. This option is incorrect as isolation of RCIC-MO-16 is initiated by a trip of the A logic channel. A candidate may choose this option if he knows the inboard and outboard isolation valves are closed by different channels but doesn't know which.

c. This option is incorrect because only the valve associated with the B logic, RCIC-MO-15, is going to close. The minimum flow valve opens if closed with the system operating and an isolation signal is present on the A side. A candidate may choose this option if he believes that this single switch operates both valves. A candidate who does not understand the isolation logic may choose this answer believing that a trip of the B logic causes a trip of the A logic.

d. This option is incorrect because the RCIC-MO-15 valve closes. The minimum flow valve is already closed but its closure on an isolation signal comes from the A logic. But a candidate may believe that both isolation actions occur. This answer is plausible because the valves do close on a full isolation signal with the system running.

Technical Reference(s):
COR002-03-02 Rev 28, Containment
COR0021802 Rev 22, Reactor Core Isolation Cooling
Procedure 2.1.22, Rev 58 Recovering From A Group Isolation

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-03-02
24. Predict the consequences of a malfunction of the following on PCIS:
d. Containment/process instrumentation

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7

Difficulty 3

ES-401 Written Examination Question Worksheet Form ES-401-5

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 239002	K3.03	
Importance Rating	4.3	

239002 SRVs

K3 Knowledge of the effect that a loss of Relief/Safety Valves will have on the following:

K3.03 Ability to rapidly depressurize the reactor.

Question: 43

With the MSIVs closed and the reactor at normal operating pressure, what is the MINIMUM number of SRVs that must be able to be opened to emergency depressurize the reactor without using alternate emergency depressurization systems?

a. 2

b 3

c. 4

d 6

Answer:

c. 4

Explanation (Why distractors are incorrect and why correct answer is correct):

The ability to rapidly depressurize the reactor is dependent on the status of the Main Steam system and the SRVs. In the event the MSIVs are open and the Main Condenser is available, depressurization is possible by manual actuation of the Main Turbine Bypass valves. If the MSIVs are closed, the speed at which the reactor can be depressurized depends on the number of SRVs that can be opened. The EOPs call for 6, if possible, and a minimum of 4 SRVs open before alternate depressurization methods are required, stating that 4 SRVs are sufficient to rapidly depressurize the RPV and maintain it in a depressurized condition. As above, any event that would prevent the opening of an SRV will adversely affect the ability to rapidly depressurize the reactor.

Distractors are plausible as each will reduce reactor pressure if the reactor is shut down.

a. This option is incorrect because 2 SRVs is not sufficient to depressurize the reactor and avoid the use of alternate depressurization systems. But since pressure control following a scram is often with the 2 Low Low Set SRVs a candidate may believe that 2 SRVs is sufficient to depressurize the reactor.

b. This option is incorrect because 3 SRVs is not sufficient to depressurize the reactor and avoid the use of alternate depressurization systems. But a candidate may believe that 3 is sufficient based on there understanding of simulated plant response where 3 SRVs do in fact lead to a significant rate of pressure reduction.

d. This option is incorrect because 6 is not the minimum but is the normal number of SRVs used to emergency depressurize which is why a candidate may choose this answer.

Technical Reference(s):
OPS Nuclear Pressure Relief / COR002-16-02 Rev 16 page 33-34

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR002-16-02;

7. Given a specific NPR malfunction, determine the effect on any of the following:
   b. Ability to rapidly depressurize the reactor

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.7 / 45.4
Difficulty	3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 259002	K5.03	
Importance Rating	3.1	

259002 Reactor Water Level Control

K5: Knowledge of the operational implications as they apply to Reactor Water Level Control system:

K5.03 Water Level Measurement

Question: 44

The plant is operating at full power and Reactor Vessel Level Control (RVLC) system operating in the automatic mode in 3-element control when validation of the level signals from NBI-LT-52A and NBI-LT-52C fail.

What is the current operating mode of the Reactor Vessel Level Control system?

a. Manual Mode

b. Master Manual Mode

c. Automatic Single-element Mode

d. Automatic 3-element Mode

Answer:

d. Automatic 3-element Mode

Explanation (Why distractors are incorrect and why correct answer is correct):

When in Auto, the RVLC system will transfer from three-element mode back to single-element mode, and prohibit transferring to three-element mode, if the conditions to satisfy three-element mode cannot be met. Two outputs provide indication on whether the Tricon controller is operating in single or three-element mode. The system will shift back to three-element mode when the requirements for three-element are met and the SW-S2 switch has been cycled back to auto (i.e., three-element).

Three element control can only be established when all of the permissives required for three- element operation are satisfied. These permissives include:

1. Total Steam Flow Good (2 or more valid steam flow transmitters)
2. At least 1 RFW Flow Measurement for each RFP is valid when both RFPTs are in Auto
3. Either Feed Pump in Auto
4. The RVLCS is in Auto

5. Total Main Steam Flow is greater than 10% of rated flow (~1.0 x 10^6 lbm/hr)
6. Reactor Vessel Level Good (1 or more valid transmitters)
7. Three Element Selector Switch RFC-SW_S2 is in AUTO

Transfer to and from three-element control occurs as permissives are met or lost. Switching to single element is automatic and transparent to the operator, except for the logic initiated alarm. It is possible to lock the system into single element control via the system selector switch RFC-SW-S2.

1. Upon loss of validation of one level signal, the system will utilize the average of the three remaining level signals.
2. Upon loss of validation of two level signals, the system will utilize the remaining two level signals.
3. Upon loss of validation of three level signals, the system will generate an alarm and use the remaining good level as the signal for control.
4. Upon loss of all valid level signals (failed, invalid or otherwise), the system will generate an alarm and revert to master manual control.
5. In the event that only two level instruments are available and they begin to diverge such that average level deviates > 4 inches from median level, the median level will become the controlling level signal. This signal will be provided by the higher of the 2 available level instruments as the logic provides a 1000 inch high level into the calculation to ensure the higher of the 2 remaining level instruments is used for the controlling RPV level signal. This condition will also result in these 2 remaining level instruments to indicate there is > 8 inches difference between these 2 instruments. A deviation alarm will be generated to Ronan and to the HMI to alert the operator of this condition. The operator will then have to determine which instrument is good, and then bypass the failing instrument in accordance with 2.4RXLVL.

a. This option is incorrect as only 2 level inputs have failed. A candidate may select this option if he knows that RVLC will shift out of automatic control on loss of level input but does not know the number of lost inputs required or if the system transfers to Manual or Master Manual.

b. This option is incorrect as only 2 level inputs have failed. A candidate may select this option if he knows that RVLC will shift out of automatic control on loss of level input but does not know the number of lost inputs required or if the system transfers to Manual or Master Manual.

c. This option is incorrect as only 2 level inputs have failed and criteria for shifting to single element are not met. A candidate may select this option if he knows that loss of input from several different signals will automatically shift control to single element but does not know the specific signals, or combination of signals, that will force the shift.

Technical Reference(s):
COR002-32-02 Rev 19, Reactor Vessel Level Control
Procedure 4.4.1, Rev 6 REACTOR VESSEL LEVEL CONTROL System

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-15-02, Rev 25

2. Describe the interrelationship between RVLC and the following:
   g. Nuclear Boiler Instrumentation

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7 / 45.3
Difficulty	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 239002	K3.03	
Importance Rating	3.7	

261000 SGTS

A1: Ability to predict and/or monitor changes in parameters associated with operating the SGTS controls including:

A1.02 Primary Containment pressure

Question: 45

Torus venting through Standby Gas Treatment (SGT) is in progress and it is noted that drywell pressure is slowly lowering. SGT-DPIC-546 is then adjusted to raise the SGT flow rate.

How does this affect the rate of drywell pressure change and why?

Drywell pressure lowers at a…

a. slower rate because of higher differential pressure across the suppression pool.

b. slower rate because of lower differential pressure across the suppression pool.

c. faster rate because of higher differential pressure across the suppression pool.

d. faster rate because of lower differential pressure across the suppression pool.

Answer:

c. faster rate because of higher differential pressure across the suppression pool.

Explanation (Why distractors are incorrect and why correct answer is correct):

2. Torus venting will decrease pressure in the torus air space, increasing the flow rate of non-condensables through the suppression pool and causing drywell pressure to decrease at a faster rate. The suppression pool acts as a loop seal between the drywell and the torus air space. Torus venting will decrease pressure in the torus air space, increasing the differential pressure across the suppression pool and increasing the flow rate of non-condensable gases from the drywell to the torus air space. Increasing the flow rate of these gases results in a faster reduction in drywell pressure.

a. This option is incorrect as drywell pressure will decrease at a faster rate. A candidate may choose this answer if he does not understand the relationship between differential pressure and the flow rate of non-condensables through the suppression pool.

b. This option is incorrect as drywell pressure will decrease at a faster rate. A candidate may

choose this answer if he does not understand the relationship between differential pressure and the flow rate of non-condensables through the suppression pool.

d. This option is incorrect as drywell pressure will decrease at a faster rate because of a higher differential pressure across the suppression pool. A candidate may choose this answer if he does not understand the relationship between differential pressure and the flow rate of non-condensables through the suppression pool.

Technical Reference(s):
2.4 PC Rev 17, Primary Containment Control
Procedure 2.2.73, R50, Standby Gas Treatment System
COR0022802 Rev 20, Standby Gas Treatment System
UFSAR Section V.2

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0022802
5. Describe the interrelationships between SGTS and the following:
b. Primary Containment

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.9/ 45.5
Difficulty	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 262001	K1.04	
Importance Rating	3.1	

262001 AC Electrical Distribution

K1: Knowledge of the physical connections and/or cause effect relationships between AC Electrical Distribution and the following:

K1.04 Uninterruptible power supply

Question: 46

The plant is shutdown when MCC-L is inadvertently de-energized. MCC-L is re-energized within a few minutes. Five minutes after the loss of MCC-L, 12.5 kV panel MDP-2 is de-energized.

What now supplies power to the Plant Management Information System (PMIS) Uninterruptible Power Supply (UPS)?

a. MCC-L

b. MDP-1

c. PMIS Batteries

d. 250 VDC bus 1A

Answer:

c. PMIS Batteries

Explanation (Why distractors are incorrect and why correct answer is correct):

Power to PMIS UPS Main panel is normally supplied from the 12.5 KV distribution system at the multi-purpose facility (MPF) panel MDP-2 through an automatic transfer switch to a battery charger. The battery charger's DC output is sent to a 75 KVA inverter and supplies a trickle charge to the battery. The inverter converts the DC voltage to AC and supplies PMIS UPS main panel. An alternate source of power from MCC-L is also provided. If the normal supply is lost, the transfer switch will shift to MCC-L unless it has been de-energized within the last 15 minutes. If it has, then the PMIS batteries supply the inverter.

a. This option is incorrect although MCC-L is the Alternate supply to PMIS/UPS it is locked out for 15 minutes after it is lost. But a candidate may choose this answer knowing that MCC-L is available and that it is the usual alternate power supply, if they forget it is locked out for 15 minutes after de-energization.

b. This option is incorrect because MDP-1 only supplies PMIS when there is an inverter failure or low battery/DC supply voltage, or an overcurrent condition on the inverter, which none of these have occurred. Long term after the batteries are depleted this may supply PMIS but the question asked what supplies it now. But since it is a power supply to PMIS a candidate may believe that in this case with MCC_L locked out and MDP-2 de-energized that this would now supply PMIS.

d. This option is incorrect. 250 VDC bus 1A is the normal supply for the No Break Power Panel, another UPS system. A candidate may choose this answer if he does not know the correct supply but remembers that 250 VDC bus 1A supplies an inverter for a system with auto transfer capabilities.

Technical Reference(s):
COR001-01-01 Rev 42, AC Electrical Distribution System

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR001-01-01 Rev 42, AC Electrical Distribution System

6. Describe the interrelationship between the AC Electrical Distribution System and the following:
   d. PMIS/UPS

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41.7
Difficulty	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 262001	2.1.27	
Importance Rating	3.9	

262001 AC Electrical Distribution

2.1.27 Knowledge of system purpose and/or function

Question: 47

What is a purpose of the 69 kV distribution system?

a. Provides one of the two AC power sources that supply the 12.5 kV distribution system.

b. Provides an alternate source of power to the 161 kV line (Auburn line) when transformer T2 is unavailable.

c. Provides AC power to emergency station auxiliaries when the generator is shutdown and start-up AC power sources are unavailable.

d. Provides the normal source of off-site AC power to the entire auxiliary power distribution system during startup and shutdown.

Answer:

c. Provides AC power to emergency station auxiliaries when the generator is shutdown and start-up AC power sources are unavailable.

Explanation (Why distractors are incorrect and why correct answer is correct):

The emergency off-site 69 KV AC power source provides AC power to emergency station auxiliaries. It is used to supply emergency station auxiliary loads only when the Main Generator is shutdown and the start-up (off-site) AC power source is unavailable.

a. This option is incorrect because the 69 kV power source does not provide power to the 12.5 kV system that power is provided by the 345 or 161 kV systems through transformers T2 and T5. But since it is supplied by either of two offsite sources a candidate may believe that the 69 kV line is one of those sources and choose this answer.

b This option is incorrect because the 69 kV line does not provide power to the 161 line to Auburn, the 161 kV line does provide an alternate source to the startup transformer so a candidate who confuses these concepts and would choose this answer.

d. This option is incorrect as this function would be from the startup transformer not the emergency transformer (69 kV line). But if a candidate confused the function of the emergency transformer with the startup they would choose this answer.

Technical Reference(s):
COR001-01-01 Rev 42, AC Electrical Distribution System

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR001-01-01 Rev 42, AC Electrical Distribution System

1. State the purpose(s) of the following AC Electrical Distribution Subsystems:

   a. 345 KV and 161 KV distribution system
   c. 4160V distribution system
   e. 120/240V AC Instrument power system
   j. 69 KV distribution system

Question Source: Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content: 41.7

Difficulty: 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 262002	K4.01	
Importance Rating	3.1	

262002 UPS (AC/DC)

K4: Knowledge of Uninterruptable Power Supply (AC/DC) design feature(s) and/or interlocks which provide for the following:

K4.01 Transfer from preferred power to alternate power supplies

Question: 48

What is a condition that results in an automatic transfer of the No-Break Power Supply from the Normal to Alternate power supply?

a. Inverter supply low current

b. Inverter output over current

c. Inverter supply over voltage

d. Inverter output over frequency

Answer:

d. Inverter output over frequency

Explanation (Why distractors are incorrect and why correct answer is correct):

Power to the No-Break Power Panel (NBPP) #1 is normally supplied from 250 VDC bus 1A through inverter 1A and a static switch. An automatic transfer from the inverter to the AC supply will occur on any of the following:
1) the inverter fails (low battery/DC supply voltage, overcurrent)
2) the inverter is turned off,
3) an overvoltage or undervoltage of ± 10%,
4) an overfrequency or underfrequency of ± 2 cycles.

An emergency (alternate) AC power source for the NBPP #1 is provided from MCC-R through a step-down transformer in the event that inverter 1A fails.

a. This option is incorrect because low inverter supply current does not cause the transfer to occur. But a candidate may believe that low supply current could signal a problem and therefore trigger the transfer to alternate supply.

b. This option is incorrect because the inverter does not transfer on an over current condition on the output side of the inverter. A candidate may choose this option if he does not know

the actual transfer setpoints but recognizes the 15% over current as a common over current trip setpoint.

c. This option is incorrect because an inverter supply overvoltage does not cause the transfer to occur. Only on the inverter output over voltage will it transfer. A candidate may choose this answer if they confuse the fact that the inverter trips on high output voltage. But this option is input and therefore does not cause the transfer.

Technical Reference(s):
COR001-01-01 Rev 42, AC Electrical Distribution System

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR001-01-01 Rev 42, AC Electrical Distribution System

9. Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following:
   c. Automatic bus transfer

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.7

Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401-5**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 263000	K2.01	
Importance Rating	3.1	

263000 DC Electrical Distribution

K2: Knowledge of electrical power supplies to the following:

K2.01 Major DC loads

Question: 49

What directly supplies power to the LPCI RHR Injection Valve (MO-25A) and Reactor Recirculation Discharge Valve (MO-53A)?

a. 250 VDC SWGR 1A

b. 250 VDC SWGR 1B

c. 250 VDC Division 1 Reactor Building Starter Rack

d. 250 VDC Division 2 Reactor Building Starter Rack

Answer:

c. 250 VDC Division 1 Reactor Building Starter Rack

Explanation (Why distractors are incorrect and why correct answer is correct):

250 VDC Division 1 Reactor Building Starter Rack supplies RHR Inboard Injection Valve (MO-25A) and Recirculation Pump Discharge Valve (MO-53A). This includes power to the MOV operator, control power, and indicating lights. Distractors are plausible as each is a DC power supply used to supply critical loads and may be selected by the candidate if the correct power supply is not known.

a. This option is incorrect because it does not directly supply power to the valves but does supply power to the 250 VDC Division 1 Reactor Building Starter Rack. But since it is in the correct division and is the ultimate supply a candidate may believe that it directly powers the valves.

b. This option is incorrect because it does not directly supply power to the valves but does supply power to the 250 VDC Division 2 Reactor Building Starter Rack. It does lead to the power to the RHR and RR division 2 (MO-25B and MO-53B) which makes this answer plausible.

d. This option is incorrect because this is the supply to the opposite division. Even if a candidate determines that LPCI RHR Injection Valve (MO-25A) and Reactor Recirculation Discharge Valve (MO-53A) are division 1 this remains plausible as in these systems there is

cross-over of power supplies, just not DC supplies. So it is reasonable that a candidate could believe this is the power supply.

Technical Reference(s):
Procedure 2.2.24.1, Rev 13 250 VDC ELECTRICAL SYSTEM (DIV 1)

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0020702, Rev 30
6. Describe the interrelationship between the DC Electrical Distribution System and the following:
   f. Reactor Recirculation system
   g. RHR

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.7

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 264000	K6.02	
Importance Rating	3.6	

264000 EDGs

K6: Knowledge of the effect that a loss or malfunction of the following will have on the Emergency Generators:

K6.02 Fuel oil pumps

Question: 50

The plant is in an outage in Mode 5 with MCC-S de-energized for inspection and cleaning when the following occur:

- Off-site power is lost.
- Both Diesel Generators start and supply their respective busses.

DG Fuel Oil Transfer Pump 1A trips and cannot be started.

How long can the diesel generators continue to operate, if at all, if these conditions persist?

Diesel Generators…

a. operate ≥ 0 hours and ≤ 3 hours.

b. operate > 3 hours and ≤ 24 hours.

c. operate > 24 hours and ≤ 48 hours.

d. operate for approximately 7 days.

Answer:

b. operate > 3 hours and ≤ 24 hours.

With the failure of the fuel oil transfer pump 1A and the unavailability of MCC-S there is no longer any method available to transfer fuel oil to the DG day tanks. So only the capacity of the fuel oil day tanks remain. At full load there is in excess of 3.9 hours of fuel and in the condition specified the DGs would be at less than full load. A Fuel Oil Day Tank is provided for each Diesel Generator. Each day tank holds approximately 2,500 gallons of fuel, approximately 2,200 gallons of this is usable. This is enough fuel to exceed the 3.9 hour full load operational requirement of Appendix R

Explanation (Why distractors are incorrect):

a. This option is incorrect because the diesel generators will continue to operate because

they are supplied by their booster pumps from the fuel oil day tank. The candidate who believes that the fuel oil transfer pumps directly supply the DG would choose this answer. (the DG is supplied directly with fuel oil from its booster pump that takes suction on the day tank)

c. This option is incorrect because the DG cannot continue to operate for this long on only the fuel oil day tank. The candidate who is unaware of the capacity of the day tank may choose this answer.

d. This option is incorrect because the DG cannot continue to operate for this long on only the fuel oil day tank. But a candidate may confuse the design capacity of the day tank with that of the storage tank which is to be able to supply one DG for 7 days and may choose this answer.

Technical Reference(s):
Procedure 2.2.20, Rev 89 STANDBY AC POWER SYSTEM (DIESEL GENERATOR)

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0020802, Rev 29
11. Predict the consequences a malfunction of the following would have on the Diesel Generators:
    b. Fuel Oil pumps

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7 /45.7
Difficulty	3

ES-401 **Written Examination Question Worksheet** **Form ES-401-5**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 264000	2.1.1	
Importance Rating	3.8	

264000 EDGs

2.1.1: Knowledge of conduct of operations

Question: 51

A Control Room Operator is supervising a Licensed Operator trainee in performance of a diesel generator operability surveillance on DG1. DG1 is currently operating in parallel with the grid when annunciators C-1/E-3 DIESEL GEN 1 TROUBLE and C-1/F-3 DIESEL GEN 1 EXCESSIVE VIBRATION alarm.

The following DG1 indications are present:

- DG1 load is at 2,000 kW.
- DG1 Frequency is at 60 Hz.
- DG1 reactive load is +200 kVAR.

What first action is required and may the trainee perform the required action under supervision of the Control Room Operator?

a. Place DIESEL GEN STOP/START switch to STOP. The trainee may perform this action.

b. Place DIESEL GEN 1BKR EG1 control switch to TRIP. The trainee may perform this action.

c. Place DIESEL GEN STOP/START switch to STOP. The trainee may **not** perform this action.

d. Place DIESEL GEN 1 BKR EG1 control switch to TRIP. The trainee may **not** perform this action.

Answer:

c. Place DIESEL GEN STOP/START switch to STOP. The trainee may **not** perform this action.

Explanation (Why distractors are incorrect and why correct answer is correct):

Annunciator Procedure 2.3_C-1 PANEL C - ANNUNCIATOR C-1 (C-1/F-3) Specifieds that this high vibration condition on the DG1 is a trip condition and indications are that the DG is continuing to run as evidenced by the load on the DG so the automatic action has failed. Conduct of operations procedure directs the operator to take action to correct a failed automatic

action. In this case the high vibration condition is a trip condition that did not occur so the appropriate action is to shutdown the DG. Which may be accomplished from the Control Room because the DG was started from the Control Room.

Additionally in Procedure 2.0.3 Conduct of Operations it states that "In the event of a plant transient or **equipment operation** not as expected, the trainee shall turn over operation of the equipment to the supervising watch-stander who shall take any necessary actions to stabilize the plant. Therefore the trainee may not perform this action.

a. This option is incorrect as the trainee is not allowed to perform this function. A candidate who understands that DG shutdown is required but who believes this action may be performed by the trainee would choose this answer. A candidate may believe that this abnormal action may be performed by the candidate as is done for normal evolutions.

b. This option is incorrect because the high vibration requires the DG to be shutdown and it continues to operate and the DG shutdown is to be performed by the Control Room Operator. If a candidate fails to evaluate the conditions they may believe the DG is tripped and that the failure is a failure of EG1 to trip. But the DG remains loaded so it has failed to trip and the required action is to shutdown the DG as that automatic action failed.

d. This option is incorrect because the high vibration requires the DG to be shutdown and it continues to operate. If a candidate fails to evaluate the conditions they may believe the DG is tripped and that the failure is a failure of EG1 to trip (the DG is motoring) But the DG remains loaded so it has failed to trip and the required action is to shutdown the DG as that automatic action failed.

Technical Reference(s):

Procedure 2.0.3 Rev 82, Conduct of Operation
Procedure 2.3_C-1, Rev 30 PANEL C - ANNUNCIATOR C-1

Proposed references to be provided to applicants during examination: None

Learning Objective:
INTO320103
C. Procedure 2.0.3, Conduct of Operations
1. Discuss the following as described in Conduct of Operations Procedure 2.0.3, Conduct of Operations:
d. Roles and Responsibilities During Training

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.10 / 45.13
Difficulty	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 300000	K1.06	
Importance Rating	3.1	

300000 Instrument Air

K1: Knowledge of connections and/or cause effect between Instrument Air system and the following:

K1.06 MSIV air

Question: 52

The plant is shutdown with an outage in progress and the drywell is open.

In this plant condition, what is supplying the pneumatic pressure to the air cylinders that operate the Inboard MSIVs (A080A-D)?

a. Nitrogen with no backup supply.

b. Nitrogen with Instrument air backup.

c. Instrument air with Nitrogen backup.

d. Instrument air with no backup supply.

Answer:

d. Instrument air with no backup supply.

Explanation (Why distractors are incorrect and why correct answer is correct):

The motive force for opening the MSIVs is pneumatics and closing force is supplied by pneumatics and closing springs. Normal steam flow tends to close the valve and higher inlet pressure tends to hold the valve closed. The valves fail closed on a loss of pneumatics. The pneumatic supply for the inboard MSIVs is nitrogen with a backup supply from Instrument Air; however, the Instrument Air supply is isolated after the Primary Containment is inerted. For personnel protection, the nitrogen is isolated and tagged during periods when drywell is open. The pneumatic supply for the outboard MSIVs is Instrument Air.

Distracters:

a. This option is incorrect because with the plant shutdown and the drywell open the containment is de-inerted so instrument air supply is available and nitrogen is isolated. This option is plausible because this would be the correct answer had the initial conditions been the primary containment inerted.

b. This option is incorrect because with the plant shutdown and the drywell open the containment is de-inerted so instrument air supply is available and nitrogen is isolated. This is plausible because nitrogen is the normal supply most of the time. If the candidate did not recall that nitrogen was isolated and controlled with a tagging order for personnel protection, he may select this option.

c. This option is incorrect because with the plant shutdown and the drywell open the containment is de-inerted so instrument air supply is available and nitrogen is isolated. If the candidate did not recall that nitrogen was isolated and controlled with a tagging order for personnel protection, he may believe nitrogen is a backup supply and choose this option.

Technical Reference(s):
COR0021402, Rev 22, Main Steam

Proposed references to be provided to applicants during examination: None

Learning Objective(s):
COR0021402, Rev 22, Main Steam
3. Describe the interrelationships between the Main Steam system and the following:
   f. Plant Air systems/N2 systems

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.4 / 45.7 to 45.8

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	1	
K/A # 400000	K6.06	
Importance Rating	2.9	

400000 Component Cooling Water

K6: Knowledge of the effect that a loss or malfunction of the following will have on the CCWS:

K6.06 Heat exchangers and condensers

Question: 53

The plant is operating at full power with Reactor Equipment Cooling (REC) Heat Exchanger A in service and (REC) Heat Exchanger B out of service for cleaning. Service Water (SW) flow lowers from 2,000 gpm to 500 gpm due to SW FCV-451A malfunction and REC temperature is 95°F and slowly rising.

Other than reducing reactor power, what action removes the largest heat load on the REC system?

a. Remove RWCU from service.

b. Shutdown all station air compressors.

c. Isolate REC to Fuel Pool Cooling heat exchangers.

d. Close REC-MO-1329, Augmented Radwaste Supply.

Answer:

a. Remove RWCU from service.

Explanation (Why distractors are incorrect and why correct answer is correct):

Per procedure 5.2REC, section 4.9 loss of SW to a single HX

4.9.3 IF REC HX outlet temperature approaches 98°F, THEN reduce REC heat load with one or both of following:

4.9.3.1 Reduce reactor power, as necessary, to maintain REC HX outlet temperature to ≤ 98°F per Procedure 2.1.10.

4.9.3.2 Rapidly remove RWCU from service per Procedure 2.2.66.

b. This option is incorrect as air compressors can be cooled by TEC. A candidate may select this option if he confuses the action to ensure at least one air compressor is being cooled by REC if TEC is lost.

c. This option is incorrect as REC to FPC is not isolated. A candidate may choose this option if he confuses this with maintaining REC flow to the FPC heat exchangers as a heat sink on a loss of Service Water.

d. This option is incorrect as isolating REC to AOG is performed on loss of REC header pressure. A candidate may choose this option if he confuses the required actions for actions taken on a loss of REC header pressure.

Technical Reference(s):
COR002-19-02 Rev 25, Reactor Equipment Cooling System
Emergency Procedure 5.2REC, Rev 16 LOSS OF REC

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-19-02
7. Predict the consequences a malfunction of the following would have on the REC system:
   a. Loss of Service Water system

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.7

Difficulty: 3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 201001	K1.09	
Importance Rating	3.1	

201001 CRD Hydraulic

K1: Knowledge of the physical connections and/or cause/effect relationships between Control Rod Drive Hydraulic System and the following:

K1.09 Plant Air Systems

Question: 54

The plant is operating when Alarm 9-5-2/G-5, SCRAM VALVE PILOT AIR HIGH PRESSURE, annunciates. A check of local scram pilot air header pressure (PI-229) indicates that actual pressure is 88 psig and steady.

How does this affect the CRDH system?

a. The scram pilot valves could unseat and leak air.

b. The scram valves take longer to open on a scram.

c. CRD flow control valve opens more causing high drive water pressure.

d. Scram inlet and outlet valves will begin to leak resulting in rods drifting into the core.

Answer:

b. The scram valves take longer to open on a scram.

Explanation (Why distractors are incorrect and why correct answer is correct):

The higher scram air header pressure will result in increased venting time on the scram air header when a scram signal is received. Since it takes longer to vent the header it will take a slightly longer time for the scram valves to open.

a. This option is incorrect because this pressure is within the operating range of the pilot valves. A candidate may select this answer if he doesn't know this pressure is within the operating range of the scram pilot solenoid valves.

c. This option is incorrect because the CRD flow control valve air supply is through a separate pressure regulator. A candidate may select this answer if he doesn't know that instrument air to the CRD flow control valve is supplied through a separate pressure regulator than the scram pilot air header.

d. This option is incorrect because this is the expected consequence of low scram pilot air header pressure. A candidate may select this answer if he recognizes that scram inlet and outlet valves leaking and rods drifting into the core is associated with scram pilot air header problems but doesn't understand leakage will occur only on low pressure.

Technical Reference(s):
COR002-04-02 Rev 23, Control Rod Drive Hydraulics

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-04-02
13. Describe the interrelationships between the Control Rod Drive Hydraulic system (CRDH) and the following:

    f. Plant Air systems

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.6 / 45.7 to 45.8

Difficulty 4

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 201003	A1.03	
Importance Rating	2.9	

201003 RMCS

A1: Ability to predict and/or monitor changes in parameters associated with operating the Control Rod and Drive Mechanism controls including:

A1.03 CRD drive water flow

Question: 55

When a control rod is being inserted what is the indicated flow through the stabilizer valves as indicated on local indicator FI-214, LOCAL DRIVE WATER HEADER FLOW INDICATOR?

a. 2 gpm

b. 4 gpm

c. 6 gpm

d. 10 gpm

Answer:

a. 2 gpm

Explanation (Why distractors are incorrect and why correct answer is correct):

The Rod Motion Control switch energizes directional control valves at the hydraulic control unit for the rod selected for motion to provide hydraulic flow to re-position the control rod. When the directional control valves energize to provide hydraulic flow to the control rod drive mechanism, the flow stabilizing valves de-energize to close to maintain a constant drive water flow rate through the CRD hydraulic system. When there is no rod movement, the flow through the insert stabilizing valve will be approximately 4 gpm. When there is an insert rod signal from the REACTOR MANUAL CONTROL SYSTEM (RMCS), the insert solenoid valve will close, balancing the 4 gpm flow directed to an HCU for normal insertion of a control rod. When there is no rod movement, the flow through the withdraw stabilizing valve will be approximately 2 gpm. A withdraw signal from RMCS will close the valve balancing the 2 gpm flow directed to an HCU for normal withdrawal of a control rod. With no rod motion in progress (both stabilizing valves open) indicated drive water flow on stabilizing valve flow indicator FI-214 will be 6 gpm. When inserting a control rod, the insert stabilizing valve closes and 4 gpm less drive water flow goes through the stabilizing valves and flow drops to 2 gpm.

b. This option is incorrect because this is the drive water flow that would go through the stabilizing valves if the control rod had been withdrawn instead of inserted. A candidate

could easily confuse which flow is associated with insert and choose this answer.

c. This option is incorrect because 6 gpm is more drive water flow than would be flowing through the stabilizing valves during insert. But this is the flow through the valves when no rod is being moved so a candidate could confuse this normal flow with insert and choose this answer.

d. This option is incorrect because 10 gpm is more than would be flowing through the stabilizing valves. But a candidate without a proper mental model of the system could pick this answer if they add the 4 gpm insert flow to the HCU to the normal flow with both stabilizing valves, 6 gpm, to get 10 gpm and choose this answer.

Technical Reference(s):
COR002-20-02 Rev 21, Reactor Manual Control System
COR002-04-02 Rev 23, Control Rod Drive Hydraulics

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-20-02 Rev 21
9. Describe the interrelationships between RMCS and/or RPIS and the following:
   a. CRDH

COR002-04-02 R23
9. Given a CRDH system component manipulation, predict and explain the changes in the following parameters:
   h. CRD drive water flow

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.5 / 45.5

Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401-5**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 290002	K3.03	
Importance Rating	3.3	

290002 Reactor Vessel Internals

K3: Knowledge of the effect that a loss or malfunction of the Reactor Vessel Internals will have on following:

K3.03 Reactor Power

Question: 56

The plant is operating at 100% power with Recirculation Pump flows matched when foreign material completely plugs jet pump 12 nozzle. (See attached figure)

When conditions stabilize, what is the effect on reactor power and how is flow indication for jet pump 12 affected?

Reactor power…

a. rises and jet pump 12 indicated flow falls to zero and remains at zero.

b. lowers and jet pump 12 indicated flow falls to zero and remains at zero.

c. rises and jet pump 12 indicated flow lowers to zero then rises above zero.

d. lowers and jet pump 12 indicated flow lowers to zero then rises above zero.

Answer:

d. lowers and jet pump 12 indicated flow lowers to zero then rises above zero.

Explanation (Why distractors are incorrect and why correct answer is correct):

Since core flow directly effects reactor power, any manipulations or malfunctions that effect flow will have the corresponding effect on reactor power. The plugging of one jet pump nozzle while at full power causes two conditions to occur.

1. Flow is reversed through the diffuser on the plugged jet pump, allowing more flow to bypass the core. This reverse flow through the jet pump will cause a flow indication for that jet pump to lower, it will still be reading flow as flow in either direction through the jet pump creates a DP and an indication of flow. It will however be less flow (absolute magnitude), less DP and less indicated flow than before.

2. The remaining 19 jet pumps will operate at a slightly higher flow ratio due to the changed hydraulic conditions (less resistance to flow).

The net effect would be a reduction in core flow to approximately 96% of rated with a corresponding decrease in reactor power.

a. This option is incorrect because reactor power lowers and indicated flow in that jet pump will indicate above zero when conditions stabilize. A candidate who recalls that flow through the remaining jet pumps rises may believe that core conditions could cause power to rise an may also reason that if the nozzle is completely plugged that flow must indicate zero.

b. This option is incorrect because following the plugging flow indication for the jet pump would stabilize at some value above zero when reverse flow is established through the jet pump. Since the jet pump flow indication is jet pump differential pressure, once reverse flow is established indication would stabilize at some level above zero. But a candidate who does not know how the jet pump flow indication works may choose this answer reasoning that with the nozzle plugged that flow indication must therefore be zero.

c. This option is incorrect because power lowers a candidate who does not understand the effect of this plugged jet pump may choose this answer because for many operational conditions power does rise, as in the case of pressure events, flow increases and cold water injection.

Technical Reference(s):
COR0011501 Rev 24, Nuclear Boiler
COR0022202 Rev 30, Reactor Recirculation

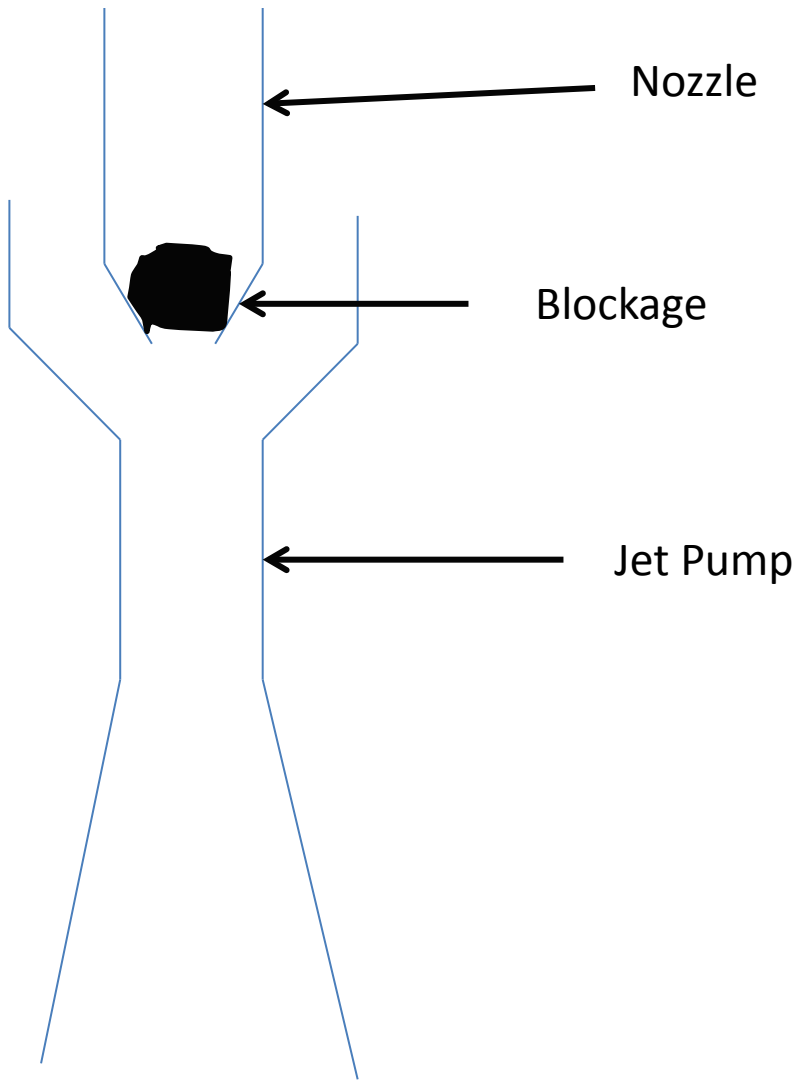
Proposed references to be provided to applicants during examination: None

Learning Objectives:
COR0011501
6. Given a specific Nuclear Boiler system malfunction, determine the effect on any of the following:
   c. Reactor Power

COR0022202
6. Given a specific Reactor Recirculation system or the Recirculation Flow Control system malfunction, determine the effect on any of the following:
   d. Reactor power (normal and reduced forced flow conditions)

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7/ 45.4
Difficulty	3

Question 56: FIGURE



ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 204000	2.1.41	
Importance Rating	2.8	

204000 RWCU

2.1.41 Knowledge of the refueling process.

Question: 57

The plant is shutdown with refueling in progress and Feedwater line A is isolated. What is the purpose of establishing the cross-tie between Reactor Water Cleanup (RWCU) and Fuel Pool Cooling (FPC)?

a. Provide alternate decay heat removal.

b. Provide a return path for the RWCU system.

c. Provide a drain path to prevent overfilling the fuel pool.

d. Provide an alternate method to makeup to the fuel pool.

Answer:

b. Provide a return path for the RWCU system.

During MODE 5 operation with FW line A out of service, the normal RWCU return path to the reactor vessel is isolated. The RWCU system can be cross- tied with the Fuel Pool Cooling (FPC) system establishing a return flow path to the reactor cavity. A spool piece must be installed allowing RWCU return to be aligned to the Fuel Pool Cooling system recirculation line.

Explanation (Why distractors are incorrect and why correct answer is correct):

a. This option is incorrect because although RWCU does provide cooling the purpose here is to allow its operation of the system to provide a return flowpath. A candidate may choose this answer if he understands that RWCU operation during an outage facilitates decay heat removal from the fuel, and does not consider that RHR and/or the Alternate Decay Heat Removal System (ADHRS) would be used for emergency heat removal.

c. This option is incorrect because installing the spool piece is not necessary for this function. A candidate may choose this answer if he understands that a drain path through RWCU may be used to lower level in the reactor cavity but does not understand that a cross-tie to FPCCU is not necessary to establish a flow path.

d. This option is incorrect because the purpose of the cross-tie is to allow a return flow path to RWCU not makeup to the fuel pool. But since fuel pool level is an operational concern a candidate could believe that this is the purpose.

Technical Reference(s):
COR001-20-01 Rev 18, Reactor Water Cleanup
COR001-06-01 Rev 26, Fuel Pool Cooling
Procedure 2.2.66 Rev 103, Reactor Water Cleanup

Proposed references to be provided to applicants during examination: None

Learning Objective(s):
COR001-18-01 Rev 24
4. Briefly describe the interrelationship between the RWCU system and the following
   s. Fuel Pool Cooling system

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.3 / 43.6

Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401-5**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 216000	A2.04	
Importance Rating	2.9	

216000 Nuclear Boiler Inst.

A2: Ability to (a) predict impacts of the following on the Nuclear Boiler Instrumentation and (b) based on these impacts use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.04 Detector diaphragm failure or leakage

Question: 58

The plant is operating at 100% power with NBI-LT-52A (Narrow Range Reactor Water Level) and NBI-LT59D (Wide Range Reactor Water Level) failed and bypassed. Subsequently a small diaphragm leak occurs on NBI-LT-52B (Narrow Range Reactor Water Level).

How is **actual reactor water level affected,** and what action is appropriate?

Actual reactor water level…

a. rises; reduce reactor power.

b. lowers; reduce reactor power.

c. rises; place MASTER LEVEL controller in MAN.

d. lowers; place MASTER LEVEL controller in MAN.

Answer:

d. lowers; place MASTER LEVEL controller in MAN.

Explanation (Why distractors are incorrect and why correct answer is correct):

A diaphragm rupture allows the pressure between the reference and variable legs to equalize. This causes a zero difference in pressure between the two legs. Zero differential pressure indicates maximum level for the instrument.

In the initial conditions specified NBI-LT-52A (Narrow Range Reactor Water Level) and NBI-LT59D (Wide Range Reactor Water Level) are bypasses so only two level instruments are available. When the diaphragm failure occurs the two levels begin to diverge such that average level deviates > 4 inches from median level, the median level will become the controlling level signal. This signal will be provided by the higher of the 2 available level instruments as the logic provides a 1000 inch high level into the calculation to ensure the higher of the 2 remaining level instruments is used for the controlling RPV level signal. So now the high level instrument (NBI-LT-52B) is also the failed instrument it will continue to rise even as the RVLC system lowers

feedwater flow and actual level lowers. As is the case with all automatic functions when the automatic function fails the operator takes control of that process in accordance with 2.4RXLVL.

a. This option is incorrect as actual level would lower because the controlling indicated level is rising due to the diaphragm failure and power reduction would not fix the mismatch in steam and feed flow because feed flow would continue to lower as the selected level instrument rises, Manual control of the Master Controller would however allow feedwater flow to be raised. But because 2.4RXLVL has a specific step to take for lowing level, that requires power reduction a candidate may confuse the conditions where power reduction is required with these conditions and choose this answer.

b. This option is incorrect because a power reduction would not fix the mismatch in steam and feed flow because feed flow would continue to lower as the selected level instrument rises, Manual control of the Master Controller would however allow feedwater flow to be raised. But because 2.4RXLVL has a specific step to take for lowing level, that requires power reduction a candidate may confuse the conditions where power reduction is required with these conditions and choose this answer.

c. This option is incorrect because actual level lowers. A candidate who mistakenly believes that the diaphragm failure would cause low indication and who believes that that low indicating would be the controlling indication would choose this answer.

Technical Reference(s):
Procedure 2.4RXLVL, Rev 25 RPV Water Level Control Trouble
Procedure 4.4.1 Rev 6, Reactor Vessel Level Control System
ARP 9-5-2/G-4, Rev 41 RVLC SYSTEM TROUBLE

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0021502, Rev 25
5. Predict the consequences of the following on the NBI:
   d. Detector diaphragm failure or leakage

Question Source:	Bank #	
	Modified Bank #	
	New X	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41.7	
Difficulty	4	

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 272000	K2.03	
Importance Rating	2.5	

226001 RHR/LPCI containment Spray Mode

K4: Knowledge of RHR/LPCI Containment Spray System Mode design features and/or interlocks which provide for the following:

K4.12 Prevention of inadvertent containment spray activation

Question: 59

The plant is operating normally except post maintenance testing of RHR-MO-26A (Drywell Inboard Spray Valve) is required. The test requires the valve to be stroked full open.

What condition or action, by itself, allows RHR-MO-26A (Drywell Inboard Spray Valve) to be opened?

a. Neither RHR pump A or C is running.

b. RHR-MO-31A (Outboard Drywell Spray Valve) is closed.

c. RHR-MO-39A (Outboard Suppression Pool Cooling Valve) is closed.

d. CONTMT COOLING VLV CONTROL PERMISSIVE switch is momentarily positioned to MANUAL.

Answer:

b. RHR-MO-31A (Outboard Drywell Spray Valve) is closed.

Explanation (Why distractors are incorrect and why correct answer is correct):

In order to prevent an inadvertent drywell spray, the Drywell Inboard Spray Valve MO-26A(B) can only be opened (in normal operation) when the Outboard Drywell Spray Valve MO-31 is closed and no LPCI initiation signal is present. If a LPCI initiation signal is present then spray valve control is required in order to open the valve.

a. This option is incorrect as the valve is interlocked closed unless the outboard is closed and this interlock is not a function of RHR pump operation. A candidate may select this option because they may understand that the purpose of the interlock is to prevent inadvertent drywell spray but do not know how that is accomplished because they may reason that if the pumps are not in operation then inadvertent drywell spray is not possible.

c. This option is incorrect because the inboard spray valve is interlocked with the outboard spray valve not the outboard suppression pool cooling valve. But Since the outboard

suppression pool cooling valve actually isolates both the suppression pool cooling and suppression pool spray path a candidate could confuse this with the drywell spray path and believe that this would prevent inadvertent spray and choose this answer.

d. This option is incorrect because there is no LPCI signal and a LPCI initiation signal is required in order to obtain spray valve control. However during accident conditions this is the action that is required in order to open the drywell spray valves so a candidate who knows this is a required action to open the valve and spray the drywell may believe that it will allow the valve to be opened.

Technical Reference(s):
COR002-23-02 Rev 30, Residual Heat Removal System

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR002-23-02
3. Describe RHR system design feature(s) and/or interlocks which provide for the following:
   q. Prevention of inadvertent containment spray activation

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.7
Difficulty 3	

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 233000	K2.02	
Importance Rating	2.8	

233000 Fuel Pool Cooling/Cleanup

K2: Knowledge of electrical power supplies to the following:

K2.02 RHR Pumps

Question: 60

During plant operation a fault causes the loss of 4160V Bus 1G. What RHR pumps remain immediately available to start?

a. A and B

b. A and C

c. B and D

d C and D

Answer:

a. A and B

Explanation (Why distractors are incorrect and why correct answer is correct):

The loss of 4160V 1G means that only 4160V 1F remains available. So only the RHR pumps powered from 4160 1F remain immediately available. RHR pumps A and B are powered from 4160V bus 1F and RHR pumps C and D are powered from 4160V bus 1G, so only RHR pumps A and B are available.

b. This option is incorrect because RHR pump C is not available because it is powered from 4160 1G. A candidate who believes that both pumps in a loop are powered by the same division, which is often the case for many systems, would choose this answer.

c. This option is incorrect because pump D is powered from 4160 1G and is not therefore available. This option would be selected by the candidate who believes that the power supplies are divisional and then also believes that B RHR loop is powered from 4160V 1F.

d. This option is incorrect because neither C nor D pump is available. But this is a highly plausible distractor for a candidate who knows the pumps in a single loop are powered by different divisions but does not recall which division powers which pumps.

Technical Reference(s):

Procedure 2.2.69.2, Rev 86 RHR SYSTEM SHUTDOWN OPERATIONS

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022302, Rev 30

2. State the electrical power supplies to the following:
   a. RHR pump motors

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.7

Difficulty 2

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A #	241000 K6.05	
Importance Rating	3.4	

241000 Reactor/Turbine Pressure Regulator

K6: Knowledge of the effect that a loss or malfunction of the following will have on the Reactor/Turbine Pressure Regulating System:

K6.05 Condenser Vacuum

Question: 61

EG 8/4/14

70%

The plant is operating steady state at 53% power and the ~~70~~5 rod line with ongoing maintenance activities. The following indications are noted:

- AR-FR-47, SJAE AIR FLOW pens are both indicating maximum flow.
- MS-PI-72A, VACUUM and MS-PI-72B, VACUUM degrade to 22.5"Hg and remain steady.

The above condition is present for the next 6 minutes. What is required and why?

a. Scram the reactor and trip the turbine because the turbine failed to automatically trip.

b. Scram the reactor and trip the turbine because of the HP turbine blade flutter phenomenon.

c. Lower reactor power by inserting control rods to avoid operating in the stability exclusion region.

d. Raise reactor power with Reactor Recirculation pumps to raise condensate depression effects on vacuum.

Answer:

a. Scram the reactor and trip the turbine because the turbine failed to automatically trip.

Explanation (Why distractors are incorrect and why correct answer is correct):

Operation is in the 5 minute delay region of CONDP graph. The DEH logic should have tripped the main turbine after 5 minutes. The operator is required to scram the reactor and then trip the turbine.

b. The problem is low pressure turbine flutter and not high pressure turbine. The candidate who does not remember this phenomena would select this answer. This answer is plausible because turbine blade fluttering does exist as low turbine loads.

c. Lowering power will help with vacuum degradation but the turbine should have tripped so scram and turbine trip is the appropriate action. The candidate who doesn't realize the turbine should have tripped would select this answer. This answer is plausible because lowering power would aid in improving condenser vacuum.

d. Increase in condensate depression effects Condensate Pump NPSH. The candidate who confuses these two effects may select this answer. This answer is plausible because raising power would help with condensate depression.

Technical Reference(s):
COR002-09-02 Rev 16, Digital Electro-Hydraulic Control

Proposed references to be provided to applicants during examination: Graph CONDP

Learning Objective(s):

COR002-09-02
8. Predict the consequences a malfunction of the following would have on DEH Control system.
   f. Loss or Degraded Condenser vacuum

Question Source: Bank # 25369
Modified Bank #
New

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

10 CFR Part 55 Content: 41.7 / 45.7

Difficulty 3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 256000	A3.07	
Importance Rating	2.9	

256000 Reactor Condensate

A3: Ability to monitor automatic operations of the reactor condensate including:

A3.07 Feedwater heater level

Question: 62

During operation at full power the A3 feedwater heater level begins to rise and the following alarms annunciate:

- HEATER HIGH LEVEL, A-2/C-5
- HEATER HIGH LEVEL TRIP, A-2/C-6

What associated actions occur when each of the alarms actuate?

When the HIGH LEVEL alarms, the condensate dump valve…

a. is full open; at the HIGH LEVEL TRIP the Non Return Valve Trips and the steam dump valve opens.

b. starts to open; at the HIGH LEVEL TRIP the Non Return Valve Trips and the steam dump valve opens.

c. is full open; at the HIGH LEVEL TRIP the Non Return Valve opens and steam dump valve closes.

d. starts to open; at the HIGH LEVEL TRIP the Non Return Valve opens and steam dump valve closes.

Answer:

a. is full open; at the HIGH LEVEL TRIP the Non Return Valve Trips and the steam dump valve opens.

Explanation:

A high heater level trip will cause the associated heater trip solenoid valve to trip, thereby removing air pressure from the associated NRVs and steam dump valves. This allows the non-return valves to function as check valves and the steam dump valves to open. At a high level in the associated heater the condensate dump valve is fully open.

Distracters:

b. This option is incorrect because the dump valve is full open at the high level alarm. A candidate could believe that at the high level that the corrective action of opening the dump valve starts (instead of being full open) and choose this answer.

c. This option is incorrect because the Non Return Valve trips instead of opening the air is removed from the operator allowing the valve to act as a check valve and the steam dump valve opens. A candidate who is unfamiliar with the configuration of the heaters and their associated valves may believe this is the correct answer if they believe that opening the NRV would help to lower level by allowing more flow to leave the heater.

d. This option is incorrect because the dump valve is full open at the high level alarm and the Non Return Valve trips instead of opening, its operating air is removed from the operator allowing the valve to act as a check valve and the steam dump valve opens as well. The candidate that would choose this answer is unfamiliar with the configuration of the valves associated with the heater and is unfamiliar with the actions that occur on rising level.

Technical Reference(s):
Procedure 2.2.29, Rev 53 FEEDWATER HEATERS AND EXTRACTION STEAM SYSTEM
Procedure 2.3_.A-2, Rev 36 PANEL A - ANNUNCIATOR A-2

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0010401, Rev 21
2. Describe the purpose of the following major components in the extraction steam and heater drains system:
   c. Level Control Valves (LCVs)
5. Describe the extraction steam and heater drains design features and/or interlocks that provide for non-return valve operation.

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	41.7 / 45.7
Difficulty	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 259001	A4.06	
Importance Rating	3.4	

259001 Reactor Feedwater

A4: Ability to manually operate and/or monitor in the control room:

A4.06 Feedwater Inlet Temperature

Question: 63

Reactor power is at 90% and has been steady for the past two days. In the last twenty minutes RF-TE-1, Feed Water Temperature, has lowered 10 degrees.

What indication is preferred for validating this condition, given that all listed indications are operable?

a. PMIS Feedwater Heating Display (FWHEAT)

b. RR-TR-165, RR Suction and FW Temperature (Panel 9-4)

c. Average of Reactor FW Channel A1 and B1 (B030 and B031)

d. RF-TI-1, Reactor Feed Pump Discharge Header Temperature

Answer:

a. PMIS Feedwater Heating Display (FWHEAT)

Explanation (Why distractors are incorrect and why correct answer is correct):

Feedwater temperature indications in Step 4.3 <u>shall be used in the order listed</u> unless inoperable. <u>Operational decisions should not be made using lower order indications</u> (e.g., RR-TR-165) <u>if higher order indications</u> (e.g., PMIS points) <u>are available</u>.

4.3 Monitor feedwater temperature using following PMIS points (with a quality code of GOOD) or indicators:

4.3.1 PMIS FEEDWATER HEATING DISPLAY (FWHEAT Turn on Code).
4.3.2 PMIS Point NSSRP617, FEEDWATER TEMPERATURE.
4.3.3 Average of following PMIS points:
4.3.3.1 B030, REACTOR FW CHNL B1 TEMP.
4.3.3.2 B031, REACTOR FW CHNL A1 TEMP.
4.3.3.3 B032, REACTOR FW CHNL B2 TEMP.
4.3.3.4 B033, REACTOR FW CHNL A2 TEMP.
4.3.4 RF-TI-1, RFP DISCH HDR TEMP (Board A).
4.3.5 RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4).

b. This option is incorrect. RR-TR-165, RR SUCTION & FEEDWATER TEMP is not to be used if higher tiered indication is available. A candidate may select this option if he knows that it is an available indication for feedwater temperature but does not know the required hierarchy of use.

c. This option is incorrect. Averaging PMIS points is not to be used if a higher tiered indication is available, and all 4 points must be averaged, not just A1 and B1. A candidate may select this option if he knows that it is an available indication for feedwater temperature but does not know the required hierarchy of use.

d. This option is incorrect. RF-TI-1, RFP DISCH HDR TEMP is not to be used if higher tiered indication is available. A candidate may select this option if he knows that it is an available indication for feedwater temperature but does not know the required hierarchy of use.

Technical Reference(s):
INT032-01-35 Rev 5, CNS Abnormal Procedures Condensate/Feedwater
Procedure 2.4EX-STM, Rev 17

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT032-01-35
L. Given plant condition(s) and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.10
Difficulty	3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 272000	K2.03	
Importance Rating	2.5	

272000 Radiation Monitoring

K2: Knowledge of electrical power supplies to the following:

K2.03 Stack gas radiation monitoring system

Question: 64

What is the power supply to the Elevated Release Point Radiation Monitor sample pump?

a. NBPP

b. CCP1B

c. CCP2B

d. PPGB-1

Answer:

d. PPGB-1

Explanation (Why distractors are incorrect and why correct answer is correct):

PPGB-1 is listed as the power supply for the ERP sample pump.

Each of the power supplies listed is a plausible answer as each provides power to an ERP Radiation Monitoring system control or component. A candidate that is not certain of the correct power supply may remember the power supplies listed as distractors as being associated with the rad monitor.

a. This option is incorrect because this is the power supply to the ERP flow detector and since it is a related component a candidate may confuse this power with the actual power supply and choose this answer.

b. This option is incorrect because this is the power supply to the Turbine Building vent monitor and since it is a similar component a candidate may confuse this power with the actual power supply and choose this answer.

c. This option is incorrect because this is the power supply to the ERP Skid Mounted Indication and Control Unit and since it is a related component a candidate may confuse this power with the actual power supply and choose this answer.

Technical Reference(s):
Radiation Monitoring/COR001-18-01 Rev 24
Procedure 4.15.1, Rev 23 Elevated Release Point Radiation Monitoring System

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR001-18-01 Rev 24
6. State the electrical power supply to the following:
   c. Kamans

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.7
Difficulty:	3

Examination Outline Cross-Reference:

Level	RO	~~SRO~~
Tier #	2	
Group #	2	
K/A # 290001	K5.01	
Importance Rating	3.3	

290001 Secondary CTMT

K5: Knowledge of the operational implications of the following concepts as they apply to Secondary Containment:

K5.01 Vacuum breaker operation

Question: 65

The following containment conditions exist:

- Pressure in the reactor building is 14.7 psia.
- Pressure in the torus is 14.8 psia.
- Pressure in the drywell is 14.9 psia.

Standby Gas Treatment system is now aligned to vent from the drywell only. If venting continues, at what approximate drywell pressure do the reactor building to torus vacuum breakers **first** operate?

At approximately…

a. 14.1 psia

b. 14.3 psia

c. 14.6 psia

d. 14.8 psia

Answer:

a. 14.1 psia

Explanation (Why distractors are incorrect and why correct answer is correct):

The Torus-to-Drywell Vacuum Breakers (12) relieve pressure from the Torus to the Drywell if there is a pressure differential greater than 0.5 psid. This pressure differential could develop following a LOCA. As the steam released into the Drywell condenses, it could develop a vacuum in the Drywell, and the atmospheric pressure from the outside could exceed the design negative pressure limit of 2.0 psid and collapse the Drywell. The vacuum breakers will open to equalize pressure between the Drywell and Torus. **The opening setpoint is approximately 0.1 psid by design.**

The Reactor Building (secondary containment) to Torus vacuum breaker system (2 100% capacity vacuum breakers) relieves pressure from the Reactor Building to the torus if torus pressure were to drop to 0.5 psi below Reactor Building pressure. Operation of either vacuum breaker will maintain a pressure differential of less than 2 psid, the external design pressure of the primary containment.

So with reactor building pressure at 14.7 psig the reactor building to torus vacuum breakers will open when Torus pressure is at 14.2 psig. In order for the torus pressure to drop to 14.2 psig the it must relieve to the drywell and assuming an opening setpoint of 0.1 psid that would mean that as drywell pressure fell that at approximately 14.1 psia drywell pressure the RB to torus vacuum breakers would open.

b. This option is incorrect because at a drywell pressure of 14.3 psia there would not be sufficient pressure differential to open the reactor building to torus vacuum breakers even if the torus were to be equalized with the drywell which would be slightly less than the torus So at this drywell pressure the RB to torus vacuum breakers are closed. A candidate who knows the opening setpoints for the valves but then subtracts the combined DP of 0.6 from initial DW pressure and not the reactor building pressure would choose this answer.

c. This option is incorrect because at this drywell the valves would be closed. A candidate who only considers the 0.1 psid pressure opening setpoint of the drywell to torus breakers and fails to consider the additional pressure differential to open the RB to Torus breakers would choose this answer.

d. This option is incorrect because at this drywell the valves would be closed. A candidate who only considers the 0.1 psid pressure opening setpoint of the drywell to torus breakers and then subtracts this from the initial drywell pressure and not the initial RB pressure would choose this answer.

Technical Reference(s):
COR002-03-02 Rev 28, Containment
Tech Spec 3.6.1.7 Bases

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR002-03-02 R28
12. Describe the Containment design features and/or interlocks that provide for the following:
    f. Reactor Building to Torus D/P
16. Given a Containment/PCIS component manipulation, predict and explain the changes in the following:

    d. Drywell to Suppression Chamber D/P
    e. Reactor Building to Suppression Chamber D/P

Question Source: Bank #
Modified Bank #

New X

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

10 CFR Part 55 Content: 41.9 / 45.3

Difficulty 3

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:		
	Level	RO ~~SRO~~
	Tier #	3
	K/A #	2.1.18
	Importance Rating	3.6

1. Conduct of Operations

2.1.18 Ability to make accurate, clear, and concise logs, records, status boards, and reports.

Question: 66

The plant is operating at rated power when an ice dam upstream of Cooper causes river level to lower. The following occurs as a result:

- River level lowers to 873' MSL.
- Procedure 5.2SW is entered.
- At a river level of 870' MSL EPIP 5.7.1, Emergency Classification, is entered and an Unusual Event is declared.
- Service Water (SW) is declared inoperable.

What Control Room Narrative Log recording(s) shall be made by the BOP Operator?

a. Entry into 5.2SW only.

b. River Level at 873 and River Level at 870.

c. Entry into 5.2SW and declaration of SW inoperability.

d. Declaration of Unusual Event and Service Water inoperability.

Answer:

a. Entry into 5.2SW only.

Explanation (Why distractors are incorrect and why correct answer is correct):

Per Operating Instruction 20:

Following is a list of entries that shall be made in the CNS Operations Log by the SM or CRS.

NOTE – LCO Conditions momentarily entered by switch operation in accordance with COP 2.0.11 requirements, are not required to be logged

(1) Inoperable and operable equipment declarations. Entry should specify whether inoperability was planned or unplanned.

(2) When equipment is made or found to be inoperable log the initial verification or demonstration of operability of the required redundant equipment. Subsequent logging of recurring actions shall be done in Procedure 6.LOG.601or NOMs LCO tracker.

(3) Reasonable expectation of operability for equipment or an Operable-Judgment classification of a Condition Report.

(4) Changes in system availability per requirements 0.49.

(5) Risk reviews associated with switching orders/switchyard work that has not been documented through the work control process.

(6) Initial availability of indications to plant Operators that are not recorded on a recording device meeting the threshold of an EAL condition (i.e. High river/forebay water level > 899' MSL, A security condition as reported by the Security Shift Supervisor)

The following is a list of entries that shall be made in the CNS Operations Log by a CRO.

(1) Plant Mode, Reactor Pressure, Reactor Power, Recirc. Loop flows, Generator Gross Output, and Rod Sequence.

(2) Entry into an Abnormal Procedure, Emergency Procedure , EOP or EPIP.

b. This option is incorrect because there is no requirement to log river level at this point Although it may very well be logged there is not an OI-20 requirement to log these levels in part because there is a trend record of river level.

c. This option is incorrect because although the procedure entry is required to be logged the SW inoperability is required to be logged by the Shift Manager or CRS. A candidate may chose this option if he does not know the specific logging requirements but assumes significant activities would be logged by the CRO.

d. This option is incorrect because there is no requirement to log either the declaration of the unusual event (although the entry into the EPIP is however required) or the SW inoperability. A candidate may chose this option because both are significant operational events.

Technical Reference(s):
Operations Instruction 20 Rev 15, Narrative Logs

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

INT032-01-03 Rev 9
B. Procedure 2.0.2, Operations Logs and Reports
1. Discuss the following as described in Conduct of Operations Procedure 2.0.2, Operations Logs and Reports:
a. Narrative Logs

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.10 / 45.13

Difficulty 3

Examination Outline Cross-Reference:	Level	RO ~~SRO~~
	Tier #	3
	K/A #	2.1.31
	Importance Rating	4.6

1. Conduct of Operations

2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Question: 67

The Control Room discharge pressure indication for a pump is found to indicate about 100 psig higher than actual because the local transmitter at the pump discharge is out of calibration. The Instrument and Control group is preparing to recalibrate the local transmitter.

What action is appropriate to inform personnel in the Control Room that maintenance is being performed on the pump discharge indication?

a. Entering the indicator status into the Night Order Log.

b. Entering the indicator status into the Control Room Narrative Log.

c. Marking the indicator on the control room panel with a red magnetic arrow.

d. Placing a caution tag on the associated pump control switch on the control room panel.

Answer:

c. Marking the indicator on the control room panel with a red magnetic arrow.

Explanation (Why distractors are incorrect and why correct answer is correct):

7.1.3 Instrument indicators or controls that are placed in an abnormal state or require particular attention shall be designated by placement of a small magnetic base red arrow unless controlled per Procedure 0.9 or identified as deficient per Procedure 0.40.

7.1.3.1 Red arrows are for enhanced operational awareness only.

7.1.3.2 Red arrows shall be removed when the component/system is returned to normal operation.

Caution Tags and associated documents used to provide a temporary means of notifying station personnel of additional precautions or instructions that affect safe operation of station equipment, or identifying and tracking components that are <u>not</u> in their normal position and are <u>not</u> being controlled by any other station procedure or process.

a. This option is incorrect because the night order log is for information and not controlling equipment. A night order would not be appropriate for the indication out of specification. The candidate that does not understand this may select this answer. This answer is plausible because Night Orders are often used at the station.

b. This option is incorrect because entering the equipment problem in the Narrative Log may be used to track status but it does not provide a visual indication to increase operator awareness of the abnormal condition. A candidate may choose this option if he does not understand the use of the red magnetic arrows to promote operator awareness of abnormal indication or control position.

d. This option is incorrect because additional information is not necessary for the safe operation of plant equipment. A candidate may choose this option knowing that caution tags are operationally significant but not specifically their application.

Technical Reference(s):
Procedure 2.0.3 Rev 82, Conduct of Operations

Proposed references to be provided to applicants during examination: None

Learning Objective(s):
SKL0100102 Rev 16, Initial License Self-Study Program
5. OI #19, Use of Temporary Control Room Information Labels/Tags:
a. Describe the use of the various types of Temporary Control Room Labels/Tags.

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.10 / 45.12

Difficulty 3

Examination Outline Cross-Reference:	Level	RO	~~SRO~~
	Tier #	3	
	K/A #	2.2.2	
	Importance Rating	4.6	

1. Equipment Control

2.2.2 Ability to manipulate the console controls as required to operate the facility between shutdown and the designated power levels.

Question: 68

A 4160V motor has the following operating history:

0600 Motor is idle.
0800 Motor is started but then quickly stopped due to system misalignment.

At 0900 the system is realigned. What is the soonest that the motor can be started, and what is the maximum number of starts that are allowed at that time?

The motor can be started…

a. immediately, only one start is allowed.

b. immediately, and two starts are allowed.

c. at 1000, only one start is allowed.

d. at 1000, two starts are allowed.

Answer:

a. immediately, only one start is allowed.

Explanation (Why distractors are incorrect and why correct answer is correct):

Throughout the operating procedures the following are the start restrictions for all 4160V motors.

1. Observe following 4160V motor starting restrictions:

NOTE 1 – A start is closure of motor supply breaker.

NOTE 2 – A cold motor is a motor which has been idle for at least 2 hours.

NOTE 3 – A hot motor is a motor which has been energized within last 2 hours.

1.1 A motor shall be at a stop prior to a start attempt.

1.2 A cold motor shall have no more than two initial start attempts. Each additional start shall be preceded by a 1 hour wait.

1.3 A hot motor shall have no more than one initial start attempt. Each additional start attempt must be preceded by a 1 hour wait.

b. This option is incorrect as only 1 start attempt may be made before allowing a 1-hour cool down period as this is currently a hot motor. A candidate may choose this option if they believe that sufficient time has elapsed to consider this a cold motor.

c. This option is incorrect as the pump has been run within the last 2 hours and is considered a hot motor. A candidate may choose this option if they believe that the next start attempt is only allowed when the motor is cold.

d. This option is incorrect as the pump has been run within the last 2 hours and is considered a hot pump and is limited to 1 start attempt before allowing a 1-hour cool down period. A candidate may choose this option if they believe that the next start attempt is not allowed until the motor is cold and then they apply all the cold start limitations which allow for two starts.

Technical Reference(s):
Procedure 2.2.3 Rev 129, Circulating Water System
Procedure 2.2.9 Rev 75, Core Spray System
Procedure 2.2.69 Rev 95, Residual Heat Removal System
Procedure 2.2.70 Rev 69, RHR Service Water Booster Pump System
Procedure 2.2.71 Rev 116, Service Water System

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

SKL012-41-01 OPS AC Electrical Distribution

6. Comply with all related AC Electrical Distribution limits and precautions.

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.10 / 45.2
Difficulty 3	

Examination Outline Cross-Reference:	Level	RO ~~SRO~~
	Tier #	3
	K/A #	2.2.22
	Importance Rating	3.7

2. Equipment Control

2.2.12 Knowledge of surveillance procedures.

Question: 69

What position, by title, can the Shift Manager delegate to authorize performance of a surveillance test?

a. Work Week Director (WWD)

b. Work Control Operator (WCO)

c. Work Control Center Supervisor (WCCS)

d. Work Control Center Administrator (WCCA)

Answer:

d. Work Control Center Administrator (WCCA)

Per procedure 0.26, section 5

NOTE - The authority to authorize performance of the test may be delegated to the CRS or the WCCA as the SM deems necessary.

Explanation (Why distractors are incorrect and why correct answer is correct):

a. This option is incorrect because the Work Week Director is not allowed to be delegated to authorize surveillance testing. The candidate that isn't aware of the delegation allowance may select this answer. This answer is plausible because the Work Week Director does authorize work to be performed by craft personnel.

b. This option is incorrect because the Work Control Operator is not allowed to be delegated to authorize surveillance testing. The candidate that isn't aware of the delegation allowance may select this answer. This answer is plausible because the Work Control Operator does authorize tagging order placement and release.

c. This option is incorrect because the Work Control Center Supervisor is not allowed to be delegated to authorize surveillance testing. The candidate that isn't aware of the delegation allowance may select this answer. This answer is plausible because the Work Control Center Supervisor does authorize work to be performed by craft personnel.

Technical Reference(s):
Procedure 0.26, Rev 66 Surveillance Program

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

INT0320101
G. Procedure 0.26, Surveillance Program
1. Discuss the following as described in Administrative Procedure 0.26, Surveillance Program:
i. Surveillance Test Authorization

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.10 / 45.13

Difficulty 2

Examination Outline Cross-Reference:	Level	RO ~~SRO~~
	Tier #	3
	K/A #	2.2.41
	Importance Rating	3.5

2. Equipment Control

2.2.41 Ability to obtain and interpret station electrical and mechanical drawings.

Question: 70

On an electrical print for a relay are contacts operated by that relay that are designated as "a" contacts and "b" contacts. During operation this relay is normally energized.

When are the "b" designation contacts open and what position (open or closed) are the contacts on the print typically shown?

They are open when the relay is…

a. energized and are shown open on the print.

b. energized and are shown closed on the print.

c. de-energized and are shown open on the print.

d. de-energized and are shown closed on the print.

Answer:

b. energized and are shown closed on the print.

Explanation (Why distractors are incorrect and why correct answer is correct):

The auxiliary contacts associated with a relay designated as “a” contacts are closed when the associated relay is energized and are open when the relay is de-energized. The “b” contacts are the opposite in that they are closed with the relay is de-energized and are open when the relay is energized. Since contacts are typically shown with the system de-energized, the “b” contacts would be shown closed because the relay is de-energized as shown on the print.

a. This option is incorrect as the "b" designation means a contact is closed when the relay is de-energized and the print is typically shown in the de-energized state meaning the contacts are shown in the closed position on the print. A candidate who understands the difference between “a” and “b” contacts but who believes the prints are shown in the normal lineup would choose this answer.

c. This option is incorrect as the "b" designation means a contact is closed when the relay is de-energized.. A candidate who confuses the two designations would choose this answer. This would be the correct answer if the question instead had asked for the “a” contacts.

d. This option is incorrect as the "b" designation means a contact is relay is de-energized. A candidate who confuses the two designations and who believes the contacts are shown in their normal position would choose this answer.

Technical Reference(s):
SKL014-02-03 Rev 4, Electrical Drawings For Licensed Operators

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

SKL014-02-03

1. Describe the operation and identification of "a" and "b" type contacts on relays and breakers.

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.10

Difficulty 2

Examination Outline Cross-Reference:	Level	RO	~~SRO~~
	Tier #	3	
	K/A #	2.3.4	
	Importance Rating	3.2	

3. Radiation Control

2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.

Question: 71

Emergency corrective action on station equipment is necessary to protect a large population from radiation exposure.

What is the MAXIMUM EXPOSURE **or** RANGE OF EXPOSURES that the Emergency Director is procedurally allowed to authorize for a radiation worker to perform these actions?

a. 5 Rem

b. 5 to 10 Rem

c. 10 to 25 Rem

d. more than 25 Rem

Answer:

c. 10 to 25 Rem

Explanation (Why distractors are incorrect and why correct answer is correct):

Per procedure 5.7.12, Section 3.1, the Emergency Director may authorize emergency exposures under the following conditions:

3.1.1 LIFE-SAVING ACTIONS 25 REM OR MORE
- 3.1.1.1 Rescue and/or treatment of personnel with life threatening injuries.
- 3.1.1.2 Corrective activities to avoid extensive exposures to large populations.

3.1.2 CORRECTIVE OR PROTECTIVE ACTIONS 10 REM TO 25 REM
- 3.1.2.1 Providing First Aid to less seriously injured personnel or in support of life saving activities.
- 3.1.2.2 Undertaking corrective action on station equipment and systems to protect large populations from radiological exposure.

3.1.3 ALL OTHER EMERGENCY CONDITIONS 5 REM

a. This option is incorrect because this is not the maximum that can be authorized. A candidate who confuses either the allowance for a planned special exposure or the normal yearly limit would choose this answer.

b. This option is incorrect because this is not the range that the Emergency Director is allowed

to authorize in this circumstance. A candidate who just believes that the yearly 5 rem limit added to the planned special exposure of 5 rem is allowed may choose this answer.

d. This option is incorrect because levels greater than 25 Rem cannot be authorized for the actions specified. A candidate who confuses the lifesaving requirement with corrective action would choose this answer.

Technical Reference(s):

Procedure 5.7.12 Rev 15, Emergency Radiation Exposure Control
INT032-01-15 Rev 6, CNS Administrative Procedures, Radiation Protection and Chemistry Procedures

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

INT032-01-15 Rev 6

1. Discuss the following as described in Emergency Preparedness Implementation Procedure (EPIP) 5.7.12, Emergency Radiation Exposure Control:
   b. Requirements
      2) Determine dose limit restrictions for emergency radiation exposures, given activity to be performed and existing conditions (per Attachment 1).

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.12 / 43.4 / 45.10
Difficulty 3	

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:		
Level	RO	~~SRO~~
Tier #	3	
K/A #	2.3.13	
Importance Rating	3.4	

3. Radiation Control

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.

Question: 72

The plant is operating at full power when annunciator 9-4-1/C-5, OFFGAS HIGH RAD, is received.

What action(s) is/are required?

a. Reduce reactor power.

b. Verify Offgas system isolates.

c. Scram the reactor and close the MSIVs and Main Steam line drains.

d. Verify the Optimum Water Chemistry (OWC) Injection System Trips.

Answer

a. Reduce reactor power.

Explanation (Why distractors are incorrect and why correct answer is correct):

PER ARP 9-4-1/C-5

1.4 Reduce power, as necessary, to lower radiation levels per Procedure 2.1.10.

1.5 Enter Procedures 2.4OG and 5.2FUEL.

Operator actions per 2.4OG, OFF-GAS SYSTEM HIGH RADIATION:

3. OFF-GAS SYSTEM HIGH RADIATION:

3.1 Lower reactor power per Procedure 2.1.10 to prevent off-gas timer from timing out or to clear Annunciator 9-4-1/C-4, OFFGAS TIMER INITIATED.

3.2 IF off-gas isolation is immediately desired, THEN at Panel 9-02, place OFFGAS TIMER switch to CLOSE.

3.3 IF fuel cladding has <u>not</u> been confirmed lost per EPIP 5.7.1, Emergency Classification, THEN request Chemistry obtain isotopic analysis of off-gas stream.

3.4 **IF Off-Gas System automatically isolates <u>or</u> is manually isolated due to high-high radiation, THEN:**

3.4.1 **SCRAM and enter Procedure 2.1.5.**

b. This option is incorrect because at the high level the Offgas System does not automatically isolate. A candidate who confuses the high with the high high alarm may choose this option because at the high high alarm level the Offgas Timer starts and Offgas isolates.

c. This option is incorrect as this action would not be required until an Off Gas High-High radiation alarm is received. A candidate may choose this option if he confuses the actions of Off Gas High Radiation and Off Gas High-High radiation.

d. This option is incorrect as this action would not be required until an Off Gas High-High radiation alarm is received. A candidate may choose this option if he confuses the actions of Off Gas High Radiation and Off Gas High-High radiation.

Technical Reference(s):
Procedure 2.4OG Rev 21, Off-Gas Abnormal
COR0011601 Rev 25, Off Gas
ARP 9-4-1/C-5, Rev 46 OFFGAS HIGH RAD

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

COR0011601
6. Describe the interrelationships between the Off Gas system and the following:
   j. Radioactive Release

Question Source:	Bank #
	Modified Bank #
	New X
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis
10 CFR Part 55 Content:	41.12 / 43.4 / 45.9 / 45.10
Difficulty	3

Examination Outline Cross-Reference:	Level	RO	~~SRO~~
	Tier #	3	
	K/A #	2.3.7	
	Importance Rating	3.5	

3. Radiation Control

2.3.7 Ability to comply with radiation work permit requirements during normal or emergency conditions.

Question: 73

What is required when entry into a high-radiation area without an SWP is performed during emergency conditions?

a. Continuous Radiation Protection (RP) coverage for the entry only.

b. Anticipated stay time and exposure is documented on the trip ticket prior to entry only.

c. Revision of the RWP stating the reason for the emergency entry and the anticipated stay time and exposure.

d. Continuous Radiation Protection (RP) coverage for the entry and initiation of an SWP following entry with time and exposure documented.

Answer:

d. Continuous Radiation Protection (RP) coverage for the entry and initiation of an SWP following entry with time and exposure documented.

Per procedure 9.ALARA.4 section 2.10: In an emergency, Radiation Protection Technicians can provide continuous radiological protection coverage for entries into an area that would normally require a SWP. The procedure makes allowance for relaxing RWP/SWP requirements during an emergency. The SWP shall be completed following the entry to document the radiation protection measures taken and the radiation dose incurred. The CNS RP-1B, RWP Time and Dose Log, shall be used during these instances to record personnel time and dose in the SWP Area. Radiation Protection personnel shall ensure this data is entered in the Radiological Data Management System RDMS.

Explanation (Why distractors are incorrect and why correct answer is correct):

a. This option is incorrect because it does not also include the initiation of the SWP following the entry. A candidate may choose this option if they know that continuous RP coverage is required but do not know the requirements for documentation of the entry.

b. This option is incorrect because the trip ticket is not required in an emergency. This answer is plausible because normal entry requires a trip ticket. The candidate who cannot recall the emergency requirements may default to the normal entry practices.

c. This option is incorrect because no RWP revision is needed in an emergency situation. Entry into a high radiation area requires a SWP not a RWP. This answer is plausible because RWPs can be revised once it is determined the current RWP does not meet the job requirements. The candidate who does not recall the emergency entry requirements may select this answer because RWP can be easily revised.

Technical Reference(s):

Procedure 9.ALARA.4 Rev 17, Radiation Work Permits

INT032-01-15 Rev 6, CNS Administrative Procedures, Radiation Protection and Chemistry Procedures

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

INT032-01-15

E.2. Discuss the compliance and use requirements associated with RWPs.

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.12 / 45.10

Difficulty 3

Examination Outline Cross-Reference:	Level	RO ~~SRO~~
	Tier #	3
	K/A #	2.4.18
	Importance Rating	3.3

4. Emergency Procedures / Plan

2.4.18 Knowledge for the specific basis for EOPs.

Question: 74

During an ATWS EOP-7A is entered; SLC is injected; and reactor water level is intentionally lowered.

When hot shutdown boron weight is injected into the reactor, water level is raised to +3 to +54 inches per step FS/L-24.

What is the reason for raising reactor water level?

a. To establish natural circulation.

b. To exit from EOP Flowchart 7A.

c. To restore adequate core cooling.

d. To allow resetting the reactor scram.

Answer:

a. To establish natural circulation.

Explanation (Why distractors are incorrect and why correct answer is correct):
FS/L-24 - Lowering RPV water level in accordance with Step FS/L-7 suppresses reactor power by reducing natural circulation core flow. However, the reduction in natural circulation flow is also expected to reduce boron mixing efficiency. Three-dimensional scale model tests suggest that as natural circulation decreases, the injected boron will have a greater tendency to stratify in the lower plenum. While reactor power will be reduced by the reduction in flow, the in-core boron concentration may remain below the value required to actually shut down the reactor. Once the Hot Shutdown Boron Weight has been injected, Step FS/L-24 therefore restores RPV water level to reestablish natural circulation flow and distribute the boron throughout the core. This strategy minimizes the integrated containment heatup by maintaining reactor power as low as possible during the time of boron injection.

Restoration of RPV water level after injection of the Hot Shutdown Boron Weight re-establishes natural circulation flow and sweeps the boron which has collected in the lower plenum into the core region.

b. This option is incorrect because EOP-7A cannot be exited unless it is determined that the reactor will remain shutdown under all conditions without boron. A candidate may select this option believing that now that the reactor is shutdown (for hot conditions) that normal EOP-1A actions are used to control level.

c. This option is incorrect because adequate core cooling would exist prior to raising reactor water level, so this action would not restore adequate core cooling. But a candidate that believes that the intentional reduction in water level precludes adequate core cooling may believe that this is the reason for raising water level.

d. This option is incorrect because water level need not be restored in order to reset the scram. The scram can be reset by defeating interlocks in accordance with EOP support procedures. A candidate who is unfamiliar with the EOP support procedures may choose his answer believing that the scram needs to be reset in order to insert control rods.

Technical Reference(s):
INT008-06-10, Rev 24 OPS EOP Flowchart 7A - RPV Level (Failure to Scram)

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

INT080610
9. Given an EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM) step, state the reason for the actions contained in the step.

Question Source:	Bank # Modified Bank # New X
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41.10 / 43.1 / 45.12

Difficulty 3

Examination Outline Cross-Reference:	Level	RO ~~SRO~~
	Tier #	4
	K/A #	2.4.26
	Importance Rating	3.1

4. Emergency Procedures / Plan

2.4.26 Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.

Question: 75

The plant is shutdown with reactor vessel head removed and the reactor cavity flooded for refueling.

What is the current minimum complement of operations personnel required for the fire brigade and who may be assigned as the Fire Brigade Leader?

The minimum operations personnel is…

a. two and the Fire Brigade leader may **only** be Work Control Operator (WCO) or Balance of Plant (BOP).

b. three and the Fire Brigade leader may **only** be Work Control Operator (WCO) or Balance of Plant (BOP).

c. two and the Fire Brigade leader may **only** be Work Control Operator (WCO), Balance of Plant (BOP) or Reactor Operator (RO).

d. three and the Fire Brigade leader may **only** be Work Control Operator (WCO), Balance of Plant (BOP) or Reactor Operator (RO).

Answer:

d. three and the Fire Brigade leader may **only** be Work Control Operator (WCO), Balance of Plant (BOP) or Reactor Operator (RO).

Explanation (Why distractors are incorrect and why correct answer is correct):

In accordance with Conduct of Operations procedure 2.0.3 when in MODE 4 or 5, the additional following requirements shall be met: One active Licensed Operator is designated as Fire Brigade Leader and the Fire Brigade shall consist of five people, three of which shall be Operations personnel. The remaining two members may be from other departments.

Procedure 2.0.3 also specifies that the Fire Brigade Leader will hold an active RO or SRO License and that the Fire Brigade Leader is normally assigned to the WCO. If the WCO is not Fire Brigade qualified, the BOP Operator should be assigned the Fire Brigade Leader and in rare cases, the RO may be assigned the Fire Brigade Leader. If the RO is assigned the Fire Brigade Leader, the CRS shall designate an individual to assume the ATCO position if an

emergency were to require the RO to respond. The designated individual shall be briefed by the CRS.

a. This option is incorrect because three is the minimum complement from operations that is required and the RO in rare cases may also be the fire brigade leader. A candidate may believe that since the unit is in mode 5 that the operations complement is lower than in modes 1, 2 and 3. Additionally a candidate may believe that the WCO and the BOP are the only options for fire brigade leader because these are the preferred positions for that role.

b. This option is incorrect because the RO may also be assigned to the role of fire brigade leader a candidate may believe that the WCO and the BOP are the only options for fire brigade leader because these are the preferred positions for that role.

c. This option is incorrect because three operations department personnel are required on the fire brigade. A candidate could reverse the number from operations (3) and the number that may come from other departments (2) and choose this answer.

Technical Reference(s):
Procedure 0.23, Rev 70CNS FIRE PROTECTION PLAN
SKL0100102 Rev 16, Initial License Self-Study

Proposed references to be provided to applicants during examination: None

Learning Objective(s):

SKL0100102, Rev 16
20. 0.23, CNS Fire Protection Plan:

Discuss the following as described in Administrative Procedure 0.23, CNS Fire Protection Plan
ii. Fire response

INT032-01-03 OPS CNS Administrative Procedure Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training
1. Discuss the following as described in Conduct of Operations Procedure 2.0.3, Conduct of Operations:

b. Control Room and Station Shift Staffing Requirements

Question Source: Bank #
Modified Bank #
New X

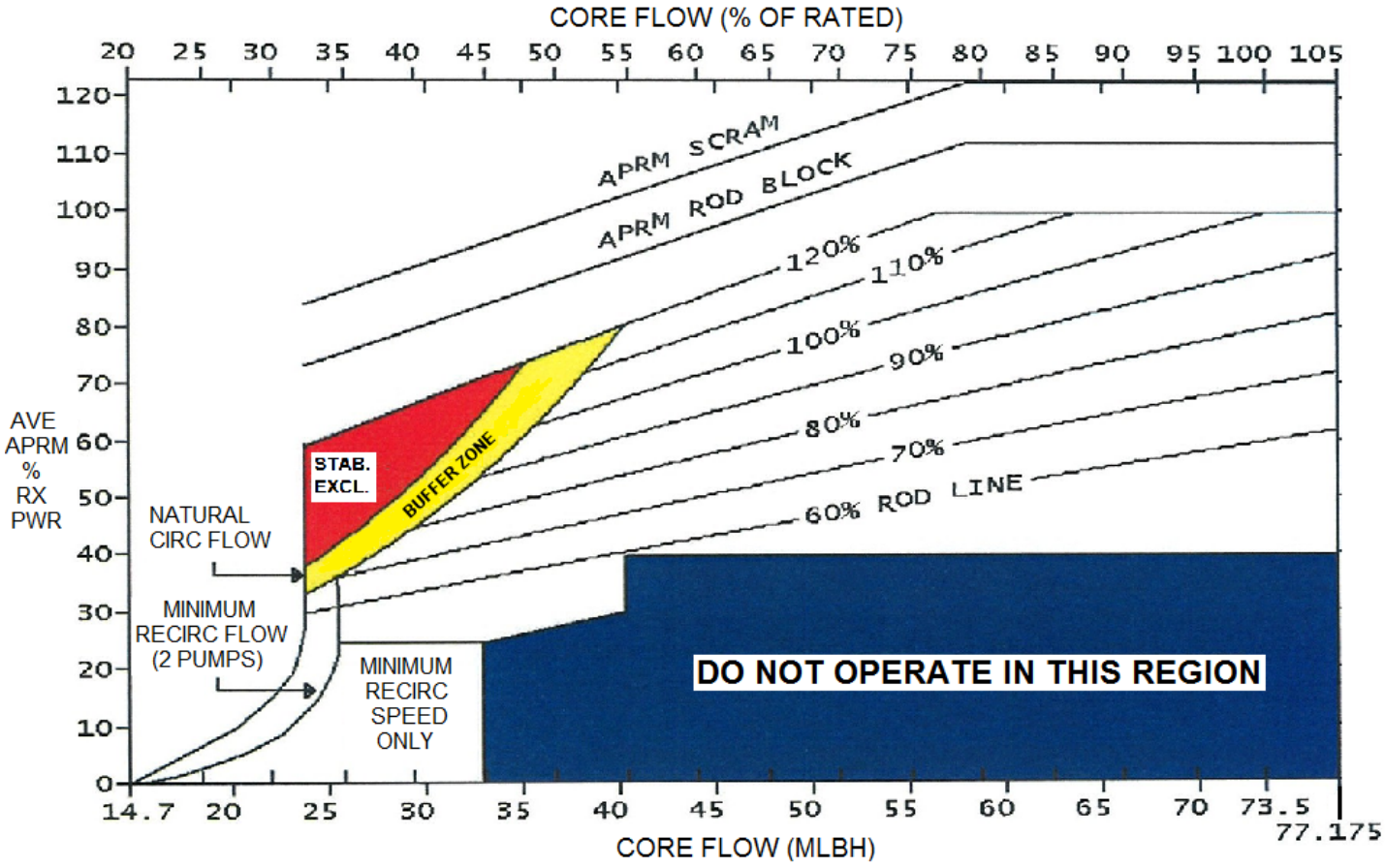
Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 41.10 / 43.5 / 45.12

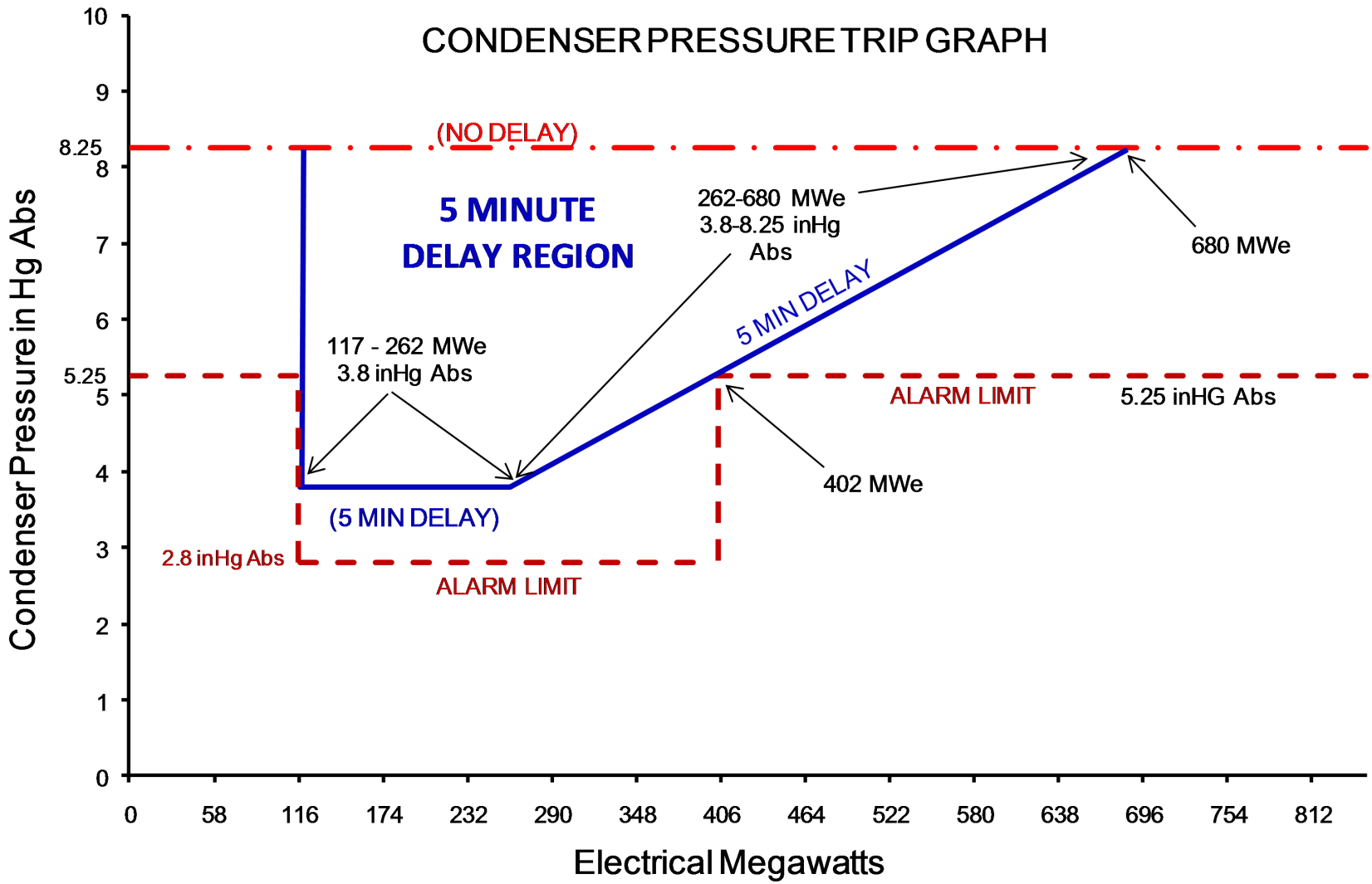
Difficulty 3

RO Handouts

POWER TO FLOW MAP - CYCLE 28



NOTE – The Maximum Effective Load Line Limit (MELLL) is 118.9% RTP (line not shown on PMIS screen).


CONDENSER PRESSURE TRIP GRAPH
(NO DELAY)
5 MINUTE
DELAY REGION
262-680 MWe
3.8-8.25 inHg
Abs
680 MWe
5 MIN DELAY
117 - 262 MWe
3.8 inHg Abs
ALARM LIMIT
5.25 inHG Abs
402 MWe
(5 MIN DELAY)
2.8 inHg Abs
ALARM LIMIT
Condenser Pressure in Hg Abs
Electrical Megawatts
10
9
8.25
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7
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5.25
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58
116
174
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290
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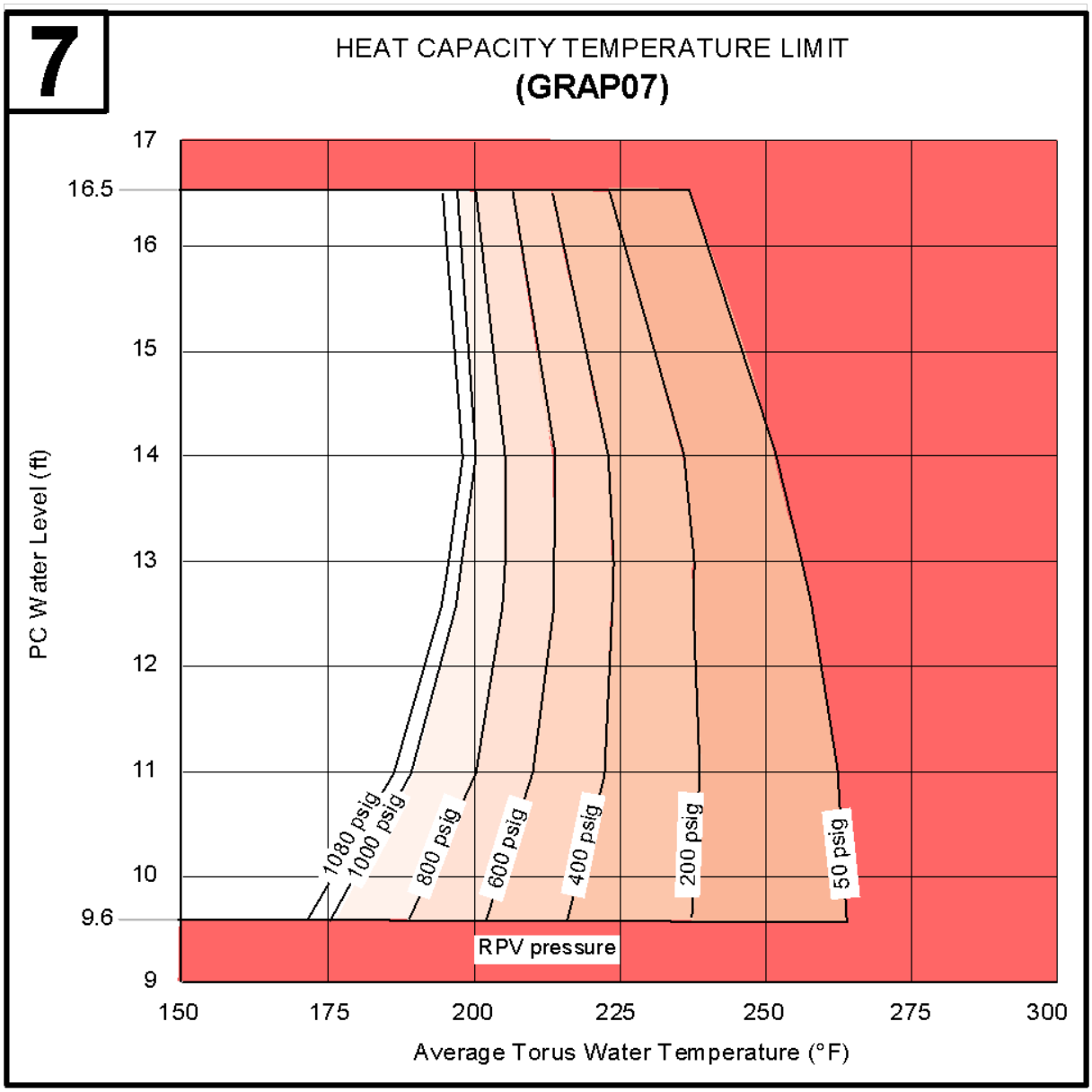
580
638
696
754
812


7
HEAT CAPACITY TEMPERATURE LIMIT
(GRAP07)
PC Water Level (ft)
17
16.5
16
15
14
13
12
11
10
9.6
9
150
175
200
225
250
275
300
Average Torus Water Temperature (°F)
1080 psig
1000 psig
800 psig
600 psig
400 psig
200 psig
50 psig
RPV pressure

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination KEY

Applicant Information

Name:	
Date: 08-04-2014	Facility/Unit: Cooper Nuclear 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 overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

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

_____
Applicant's Signature

Results

RO/SRO-Only/Total Examination Values	~~75~~ 74 K /	25 /	~~100~~ 99 K	Points
Applicant's Score	_____ /	_____ /	_____	Points
Applicant's Grade	_____ /	_____ /	_____	Percent

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		1
K/A #	295003.AA2.04	
Importance Rating		3.7

295003 Partial or Complete Loss of AC

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

Question: 76

The plant is operating at rated power during the month of July. The following alarms are received:

C-2/A-4, 480V BUS 1F MCC-L BKR TRIP
C-1/F-1, RPS PWR PANEL 1A VOLTAGE FAILURE
C-1.G-1, RPS MG SET A MOTOR DISCONNECTED

Alarm procedure actions are taken.

What is the status of the plant?
After the isolation signals are reset which procedure must the CRS enter to prioritize actions and why?

Half PCIS Group 1, 2…

a. and full Group 3 and 6.
Enter Procedure 2.2.66, RWCU SYSTEM, and restore RWCU to service to restore/maintain reactor coolant chemistry within limits.

b. and 3, and full Group 6.
Continue in Procedure 2.1.22, RECOVERING FROM A GROUP ISOLATION, and restore RRMG set ventilation before RR pumps must be tripped.

c. and full Group 3 and 6.
Continue in Procedure 2.1.22, RECOVERING FROM A GROUP ISOLATION, and restore RRMG set ventilation before RR pumps must be tripped.

d. and 3 and full Group 6.
Enter Procedure 2.2.66, RWCU SYSTEM, and restore RWCU to service to restore/maintain reactor coolant chemistry within limits.

Answer:

c. and full Group 3 and 6.

Continue in Procedure 2.1.22, RECOVERING FROM A GROUP ISOLATION, and restore RRMG set ventilation before RR pumps must be tripped.

Explanation (Optional):

The loss of MCC-L causes a loss of power to RPS MG Set A. The loss of RPS A causes a half PCIS Group 1 and 2 isolations. Per alarm C-2/A-4 and C-1/G-1 procedures, the guidance is to transfer RPS to its alternate power supply and then reset the group isolations per Procedure 2.1.22. For the Group 1 isolation, only MS-MO-74, MSL Drain Inboard valve closes. For Group 2 half the valves close and the major system effected is AOG isolation (RHR-MO-921, Steam Supply to AOG) occurs. This is not a pressing concern at the moment. A full Group 3 occurs which means RWCU isolates, so the filtration of the reactor coolant ceases. It is important to filter the reactor coolant but it is not the first priority. Procedure 8.3VIP which provides guidance on reactor coolant chemistry monitoring has a "good practice" limit on conductivity of <0.08 μmho/cm with an action point to lower power at 0.3 μmho/cm. The coolant conductivity is always kept below the "good practice" limit and there is no information in the stem that indicates there is a conductivity issue. The more pressing problem is the Group 6 isolation. The Group 6 isolation means RRMG air cooling is lost. Abnormal Procedure 2.4HVAC has a step to trip the RRMG if winding temperatures cannot be maintained below 250 deg F. Both RRMGs would be effected so both would have to be tripped which requires scramming the reactor. During July months the outside air temperature can be high enough that the ventilation must be restored in an expeditious manner because the RRMG winding temperature begins rising right away. In order to restore the ventilation quickly is to transfer RPS power to its alternate source and then reset the group isolation logic. The RRMG ventilation can be recovered in a short period using procedure 2.1.22.

Distracters:

a. This answer is incorrect because restoring RWCU is not the priority. Restoring RRMG set cooling takes precedence over RWCU restoration. The procedure listed is correct but the coolant chemistry will not appreciably be effected in the short term. The answer is plausible because reactor coolant is important to maintain within chemistry limitations. The candidate who believes this is a priority may choose this answer.

b. This answer is incorrect because a full Group 3 isolation occurs. RPS A powers the temperature switch that monitors non-regenative heat exchanger outlet temperature so both channels are tripped. The restoring RRMG ventilation is the first priority for the given conditions so this is correct. This answer is plausible because the correct action and procedure is listed. The candidate who does not recall the RPS A supply to the temperature switch may select this answer.

d. This answer is incorrect because the group isolation response is incorrect and the incorrect response is listed. A full Group 3 occurs and the filtering reactor coolant is important but not the most immediate priority. The candidate who believe reactor coolant chemistry is the most important may select this answer. This answer is plausible because the Group 6 isolation is correct.

Technical Reference(s):
Procedure 2.3_C-2, Rev. 45, Alarm tile C-2/A-4.
Procedure 2.3_C-1, Rev. 30, Alarm tiles C-1/F-1 and C-1/G-1.
Procedure 2.1.22, Rev. 58, Sections 4, 5, 6 and 9.

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR002-21-02, Rev 21
8. Given a specific RPS malfunction, determine the effect on any of the following:
   c. PCIS

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content: 55.41
55.43 5

Difficulty: 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		1
K/A #	295004 G	2.2.19
Importance Rating		3.4

295004 Partial or Total Loss of DC Pwr

2.2.19: Knowledge of maintenance work order requirements.

Question: 77

The plant is experiencing an SBO and hot shorts on a portion of the 125V DC system on the RCIC Starter Rack.

What is the Maintenance Work order requirement that governs this event and in what procedure is it located?

a. Spot Maintenance work authorization as directed in Procedure 7.0.4 CONDUCT OF MAINTENANCE.

b. Troubleshooting authorization as directed in Procedure 7.0.1.7 TROUBLESHOOTING PLANT EQUIPMENT.

c. Emergency Maintenance authorization as directed in Procedure 0-CNS-WM-102, WORK IMPLEMENTATION AND CLOSEOUT.

d. Temporary Configuration Change authorization as directed in Procedure 3.4.4 TEMPORARY CONFIGURATION CHANGE.

Answer:

c. Emergency Maintenance authorization as directed in Procedure 0-CNS-WM-102, WORK IMPLEMENTATION AND CLOSEOUT.

Explanation (Optional):
Per procedure 0-CNS-WM-102, WORK IMPLEMENTATION AND CLOSEOUT, the Shift Manager has the authority to initiate Emergency Maintenance in order to protect equipment, prevent deterioration of plant conditions and for plant personnel safety. The 125VDC system supplying RCIC, which is the only injection source available. HPCI can only be used for 10 minutes during an SBO. The hot short must be expeditiously repaired so an emergency maintenance request must be performed. (See attached procedure section).

Distracters:

a. Procedure 7.0.4 discusses maintenance and controls spot maintenance activities. The procedure does not provide emergency work. The candidate that is not completely knowledgeable about emergency maintenance would select this answer. This answer is plausible because the procedure does describe general maintenance and controls spot maintenance.

b. The existing plant condition requires immediate action be taken to correct the deficiency. The trouble shooting process is a formal pre-planned evolution requiring prior approval before commencing work. This process is not the process to use to rectify the existing condition. This answer is plausible because trouble shooting is an activity that is performed for determining the cause of plant deficiencies. The candidate that does not completely understand the trouble shooting requirement will choose this answer.

d. This procedure does provide guidance for making configuration changes to the facility on an emergent basis. It is not known with the information given in the question stem that a TCC will rectify the situation. Emergent work is required to repair the deficiency and the controlling procedure is 0-CNS-WM-102, not 3.4.4. This answer is plausible because the procedure does discuss emergency TCCs but it also states that the work control program provides guidance for implementation of the activity. The candidate that recognizes that this procedure does discuss emergent work but not the context of the work will choose this answer.

Technical Reference(s):
Procedure 0-CNS-WM-102, Rev 1 WORK IMPLEMENTATION AND CLOSEOUT

Proposed references to be provided to applicants during examination: None

Learning Objective:
SKL0110101001290A, 0.40, Work Control Program, Discuss the following as described in Administrative Procedure 0.40, Work Control Program: 1) Precautions and limitations 2) Minor maintenance 3) Emergency MWR processing.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:
55.41
55.43 5

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation.

From Procedure 0-CNS-WM-102, WORK IMPLEMENTATION AND CLOSEOUT:

7. EMERGENCY MAINTENANCE WORK ORDER PROCESSING

7.1 Under conditions, as determined by the Shift Manager, where immediate actions are required to protect the health and safety of public and plant personnel, to protect equipment or prevent deterioration of plant conditions, maintenance activities may be performed via verbal direction from the Shift Manager and on-scene Supervisor. Emergency work will be performed as outlined.

7.2 The Shift Manager shall perform the following
7.2.1 Contact Operations Supervisor and obtain verbal concurrence
7.2.2 Notify the Work Week Director or Maintenance Supervisor.
7.2.3 Make a reasonable attempt to contact the QA Manager for all EQ or Essential Work Orders.
7.2.4 Notify RP if the work will be performed in the RCA or posted area.

7.3 When the course of action is agreed upon, the Shift Manager shall verbally direct available personnel to perform the work necessary to contain or stabilize the emergency condition.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		1
K/A #	295005	G 2.4.6
Importance Rating		4.7

295005 Main Turbine Generator Trip

2.4.6 Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

Question: 78

The plant is operating at 100% power when an accident occurs that results in a **complete loss of condenser vacuum** and subsequent **Main Turbine trip**. EOP-1A is entered, CRD and RCIC are used to control RPV level and pressure. Suppression pool level begins to lower due to a leak and no suppression pool make-up sources are available.

The following indications are present:

- RPV level is 48 inches.
- RPV pressure is 920 psig.
- CRD and RCIC are running.
- Suppression pool level is at 9.8 feet and dropping at 1 inch per minutes.

What RPV pressure control action(s) is/are appropriate?

a. Anticipate emergency depressurization per EOP-1A.

b. Enter EOP-2A and discontinue actions for pressure control in EOP-1A.

c. Control pressure low but maintain RCIC as injection source per EOP-1A.

d. Enter EOP-2A and also continue actions for pressure control in EOP-1A.

Answer:

b. Enter EOP-2A and discontinue actions for pressure control in EOP-1A.

Explanation (Optional):

Following the turbine trip and loss of vacuum the condenser is no longer available for pressure control. Pressure and level are stable and suppression pool level is lowering. Because no sources of makeup water are available to the suppression pool and level is already approaching 9.6 and falling rapidly emergency depressurization is required by EOP-3A. When it is determined that emergency depressurization is required the override in EOP-1A directs that EOP-2A be entered to emergency depressurize the reactor. When this occurs the pressure control actions in EOP-1A are discontinued. The level control actions in EOP-1A however continue.

Distracters:

a. This option is incorrect because with the loss of vacuum/turbine trip the condenser is not available for anticipation of emergency depressurization. But a candidate who does not assess plant conditions and the availability of the condenser may choose this answer because with lowering suppression pool level (and volume) anticipation of emergency depressurization would be appropriate if the condenser were available.

c. This option is incorrect because although this action would be appropriate with normal suppression pool level it is not appropriate with the currently rapidly lowering SP level. The pressure control actions of EOP-1A are not used when an ED is required.

d. This option is incorrect because the pressure control actions of EOP-1A are not used during an ED. But because EOP-1A is the primary pressure control actions in other than ED situation a candidate may believe that EOP-1A actions are still appropriate.

Technical Reference(s)**:**
INT008-06-05, Rev4 24 EOP-1A
INT008-06-07, Rev 17 EOP-2A

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT00806070010600, Identify any EOP support procedure addressed in Flowchart 2A and apply any associated special operating instructions or cautions.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:
55.41
55.43 5

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation. The CRS must transition from the pressure control strategy in EOP 1A to the emergency depressurization strategy in EOP 2A.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		1
K/A #	295018.AA2.03	
Importance Rating		3.5

295018 Partial or Total Loss of CCW

AA2.03 Cause for partial or complete loss.

Question: 79

The plant is at 100% power when the reactor scrams.

- Alarm C-2/A-9, STARTUP TRANSFORMER LOCKOUT comes in.
- Drywell temperature and pressure are rising.
- Drywell moisture indication remains steady.
- 5 control rods remain at position 48.
- CRS enters EOP 1A.
- Only two REC pumps remain in operation.

What is the cause for the loss of one REC pump?
What procedure provides the appropriate guidance for the conditions given?

a Only two REC pumps are selected to STANDBY.
Remain in EOP 1A and enter Procedure 2.4CRD to restore CRD drive water.

b. Only two REC pumps are selected to STANDBY.
Transition to EOP 6A and 7A and enter Procedure 5.3EMPWR to restore REC cooling.

c. Only two REC pumps have an energized power supply.
Transition to EOP 6A and 7A and enter Procedure 5.3EMPWR to restore REC cooling.

d. Only two REC pumps have an energized power supply.
Remain in EOP 1A and enter Procedure 2.4CRD to restore CRD drive water.

Answer:

b. Only two REC pumps are selected to STANDBY.
Transition to EOP 6A and 7A and enter Procedure 5.3EMPWR to restore REC cooling.

Explanation (Optional):

On a loss of normal power, the REC pumps in STANDBY will restart when emergency power is available after a 20 second time delay. Two REC pumps are procedurally required to be in STANDBY at all times. The CRS should prioritize restoring REC for drywell and CRD pump cooling. On a reactor scram from full power, RPV water level shrink causes RPV level to lower below the EOP 1A entry level of 3 inches. Upon EOP 1A entry the SRO has to determine if the reactor will remain shutdown under all conditions without boron. The SRO must answer NO and transition to EOP 6A and 7A. The reactor is at a very low power with only 5 control rods withdrawn. There is no urgency to insert the control rods. The priority is to restore REC so the CRD pumps have cooling and then the control rods can be inserted. With the loss of drywell cooling indicated in the stem (due to a Group 6 isolation on low RPV level), REC cooling to the drywell must also be restored or else all low pressure ECCS system pumps start unnecessarily on a high drywell pressure which will put a strain on the emergency power source when the systems all start. Procedure 5.3EMPWR provides guidance for REC restoration. EOP 6A provides the direction for control rod insertion in this instance not 2.4CRD.

Distracters:

a. This answer is incorrect because the expected number of REC pumps running would be 2. Because the reactor will not remain shutdown under all conditions without boron, the SRO must transition to EOP 6A and 7A. The SRO must also enter Procedure 5.3EMPWR to mitigate the loss of power. This answer is plausible because 2.4CRD would be the correct procedure to enter to insert the control rods in any other situation where EOP 6A is not entered. The candidate who knows there is no EOP entry on the ATWS condition would know 2.4CRD is the correct procedure to insert the control rods.

c. This answer is incorrect because the number of running REC pumps is 2. The second part of the answer is correct. This answer is plausible because the correct EOP transition is listed and the correct action and procedure are correct. The candidate who does not realize the power supply to the REC pumps is the emergency supply may select this answer.

d. This answer is incorrect because the priority is to restore REC system cooling and 2.4CRD is not the correct procedure for inserting the control rods. This answer is plausible because the number of running REC pumps is correct and 2.4CRD does provide guidance for manually inserting control rods following a scram. The candidate who does not realize the transition to EOP 6A and 7A must be made may select this answer.

Technical Reference(s):
Procedure 5.3EMPWR, Rev 43
Procedure 2.4CRD, Rev 15

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT0320126Q0Q0100, Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level:

Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:

55.41
55.43 5

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation. The responsibility of the CRS is to set priorities during transients.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		1
K/A #	295025 G	2.3.14
Importance Rating		3.8

295025 High Reactor Pressure

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

Question: 80

An event occurs resulting in the following:

- RPV pressure is steady at 1080 psig with 3 SRVs open.
- No other SRVs will open.
- RPV level is -50 inches wide range and lowering 1 inch every 5 minutes.
- MAIN STM LINE HIGH RAD alarm (Annunciator 9-4-1IA-5) is sealed in.
- Drywell Radiation Monitor indicates 230 R/hr and steady.
- Drywell pressure is 2.1 psig and steady.
- RCIC is the only steam driven system and it has tripped on over speed.

The emergency plan has not yet been implemented.

What operator response is appropriate for these conditions?

a. Reset RCIC over speed trip using emergency radiological protection practices so RCIC can be placed in service.

b. Anticipate emergency depressurization and fully open main turbine bypass valves to minimize off-site release rates.

c. Reset RCIC over speed trip using standard radiological protection practices so RCIC can be placed in service.

d. Anticipate emergency depressurization and fully open main turbine bypass valves to maximize condenser plate-out.

Answer:

c. Reset RCIC over speed trip using standard radiological protection practices so RCIC can be placed in service.

Explanation (Optional):

The high RPV pressure transient has caused fuel failure. Main Steam Line Hi Radiation and high drywell radiation levels are indicative of fuel failure. With reactor pressure at 1080 psig, the reactor scrammed on high pressure. EOP 1A requires RPV pressure to be maintained below 1050 psig. Three SRVs are the only ones available and are open. Further pressure reduction is required and RCIC is the only source available for alternate pressure control. Fuel failure has caused MSIV and Drywell elevated radiation levels. There is no indication of a leak inside secondary containment so normal radiation practices are required for entering secondary containment to reset the RCIC over speed trip.

Distracters:

a. This answer is incorrect because emergency rad protection practices are not required at this point. This answer is plausible because the correct action to reset the overspeed trip is listed. The candidate that believes the radiological conditions require emergent radiological practices would select this answer.

b. With the information given, there is no reason to anticipate emergency depressurization. Reactor fuel is covered with water and containment is not threatened. Elevated secondary containment radiation levels with a primary system discharging inside secondary containment required emergency depressurization but there is no indication of a leak inside secondary containment. The elevated radiation levels are due to failed fuel. Plating out in the condenser is a good strategy to minimize off-site releases. This answer is plausible because anticipating emergency depressurization is an action in EOPs that can be performed under certain conditions.

d. With the information given, there is no reason to anticipate emergency depressurization. Reactor fuel is covered with water and containment is not threatened. Elevated secondary containment radiation levels with a primary system discharging inside secondary containment required emergency depressurization but there is no indication of a leak inside secondary containment. The elevated radiation levels are due to failed fuel. There also is not going to be a release rate that requires a General Emergency classification so emergency depressurization for this condition will not be required. The candidate that does not realize only fuel is damaged and no leak exists may select this answer. This answer is plausible because anticipating emergency depressurization is an action in EOPs that can be performed under certain conditions.

Technical Reference(s):
INT008-06-05, Rev 24 EOP 1A lesson plan

Proposed references to be provided to applicants during examination: None

Learning Objective:

INT0080605001070A State the basis for pressure control actions in Flowchart 1A as they apply to the following: Specific Setpoints

Question Source:

Bank #
Modified Bank #

New X

Question Cognitive Level:

Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content:

55.41
55.43 4

Difficulty: 3

SRO Only - 10CFR55.43 b (4radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		1
K/A #	295031	G 2.1.39
Importance Rating		4.3

295031 Reactor Low Water Level

2.1.39 Knowledge of conservative decision making practices.

Question: 81

A LOCA is ongoing and HPCI is the <u>only</u> available injection source.

- RPV pressure is 900 psig and lowering 1 psig per minute.
- RPV water level is lowering 1 inch every two minutes.
- RPV level is currently -180 inches (corrected fuel zone).

What direction does the CRS provide to the control room operator?

a. Open 6 SRVs per EOP 2A, EMERGENCY RPV DEPRESSURIZATION, and leave them open.

b. Open the main turbine bypass valves per Procedure 2.2.77.1, DIGITAL ELECTRO-HYDRAULIC (DEH) CONTROL SYSTEM, and leave them open.

c. Open 6 SRVs per EOP 2A, EMERGENCY RPV DEPRESSURIZATION, until RPV pressure lowers to 200 psig; then maintain 200 psig.

d. Open the main turbine bypass valves per Procedure 2.2.77.1, DIGITAL ELECTRO-HYDRAULIC (DEH) CONTROL SYSTEM, until RPV pressure lowers to 200 psig; then maintain 200 psig.

Answer:

c. Open 6 SRVs per EOP 2A, EMERGENCY RPV DEPRESSURIZATION, until RPV pressure lowers to 200 psig; then maintain 200 psig.

Explanation (Optional):

If RPV water level cannot be restored and maintained above -183 inches, then the CRS determines that emergency depressurization is required. Because HPCI is the only injection source available, the override in the emergency depressurization EOP flow path allows full depressurization to be ignored and RPV pressure controlled as low as practicable to maintain RPV injection. The HPCI system isolates on low RPV pressure of 150 psig. **<u>The CRS is permitted to conservatively determine emergency depressurization</u>** is required if, given the circumstances and the available systems, in the CRS' opinion that RPV level cannot be restored

above -183 inches before it lowers below that level. The CRS does not have to wait until RPV lowers below -183 inches and then try to recover level above -183 inches before deciding emergency depressurization is required. If the CRS anticipates ED, the second override on Step PC/P-1 (EOP 1A) directs opening the main turbine bypass valves per Procedure 2.2.77.1. If the CRD decides to ED, the third override on RC/P-1 (EOP 1A) sends the operator to EOP 2A to open the SRVs.

Distracters:

a. is incorrect because leaving the SRVs open causes HPCI to isolate and the loss of the only available injection source. The candidate that does not recall the override allowing partial emergency depressurization may choose this option. This option is plausible because emergency depressurization normally requires the SRVs to be opened and remain open until RPV pressure is < 50 psig below torus pressure.

b. is incorrect because leaving the BPVs open causes HPCI to isolate and the loss of the only available injection source. The bypass valves is an alternate emergency depressurization system. The SRVs are the primary source. The candidate that does not recall the override allowing partial emergency depressurization may choose this option. This option is plausible because emergency depressurization (using an alternate system) normally requires the bypass valves to be opened and remain open until RPV pressure is < 50 psig below torus pressure.

d. is incorrect because the SRVs are utilized for emergency depressurization. If no more than 4 SRVs can be opened and RPV pressure remains > 50 psig above torus pressure, then alternate systems (bypass valves is one system) can be used. The candidate the does not recall that the bypass valves are an alternate method may select this option. This answer is plausible because the bypass valves system is an alternate emergency depressurization system.

Technical Reference(s):
Student Text INT008-06-07, Rev 19
Student Text SKL008-01-02, Rev 36

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT00806070010900 State the pressure control action required if full RPV depressurization will result in a loss of injection required for adequate core cooling.
SKL00801020011800 Given that a transient has occurred, determine what Operation's policy is for properly dealing with the transient, per the guidance in GOP 2.0.3, Conduct of Operations.

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level:

Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:

55.41
55.43 5

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation.

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		1
K/A #		295038.EA2.03
Importance Rating		4.3

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

Question: 82

An accident has occurred and a release of radioactive gasses must be made from the Drywell. With the wind blowing straight for Brownville, what set of atmospheric conditions would cause the **highest** dose to a small group of Boy Scouts camping at the Brownville park?

	Wind Speed	Stability Class	Precipitation
a.	15 mph	Stable	Raining in Brownville but **not** onsite
b.	5 mph	Neutral	None
c.	25 mph	Unstable	None
d.	15 mph	Stable	Raining between onsite and Brownville

Answer:

a. 15 mph Stable Raining in Brownville but **not** onsite

Explanation (Optional):
The **highest** dose to the scouts would come from a long narrow plume and precipitation in the area that would cause wash out of the plume onto the Boy Scouts. The fastest the plume gets to Brownville yet slow enough that it remains in the area and the fact that the stability class is stable, allows it to remain concentrated over the scouts for as long as possible.

Distracters:

b. The slower the plume moves the greater the dose would be to the Boy Scouts but the greater the time for decay before it gets there. Also a stability class of neutral would cause the plume to dissipate somewhat. This answer is plausible because Brownville is only a few miles from the plant and a candidate that does not understand stability classes or release rate decay time may choose this answer.

c. The faster the plume moves the shorter time over the Boy Scouts and therefore a lower dose. Also a stability class of Unstable would cause the plume to dissipate greatly and cover a large wide area. This answer is plausible because Brownville is located close to the plant and the candidate that does not understand stability classes may choose this answer.

d. The medium speed and a stable atmosphere is ideal however the rain between the plant and Brownville would cause the washout of the radionuclides before they get to the Boy Scouts. This answer is plausible because of the stable atmosphere. The candidate that does not understand rain effects on radioactive releases may choose this answer.

Proposed references to be provided to applicants during examination: None

Technical Reference(s):
Student Text GEN003-04-01, Rev 3

Learning Objective:
GEN0030401E0E0400 Describe the atmospheric conditions associated with the stability class designations.

Question Source:
Bank # 23336
Modified Bank #
New

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

10 CFR Part 55 Content:
55.41
55.43 4

Difficulty: 3

SRO Only - 10CFR55.43 b (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		2
K/A #		295007.AA2.01
Importance Rating		4.1

295007 High Reactor Pressure

AA2. Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE:

AA2.01 Reactor pressure

Question: 83

According the Technical Specification bases, the safety mode of the Nuclear Pressure Relief system SRVs are designed to protect against which event?

a. The loss of main condenser vacuum.

b. Main Generator load reject/generator trip.

c. MSIV closure with high flux actuating an RPS scram.

a. Main Turbine stop valve closure with the failure of bypass valves.

Answer:

c. MSIV closure with high flux actuating an RPS scram.

Explanation (Optional):

Per Technical Specification Bases Section 3.4.3, the most severe pressure transient which the safety function of the SRV is to protect against is closure of all the MSIVs with the reactor scram signal coming from high flux rather than MSIV valve closure signal.

Distracters:

a. This answer is incorrect because a loss of condenser vacuum is not the most severe pressure transient. The loss of vacuum causes the turbine to trip and stop valve closure will scram the reactor before the pressure/power become too severe. The candidate who does not recall the bases of the SRV safety function may select this answer. This answer is plausible because an ATWS with bypass valve failure can result in a pressure perturbation when the main turbine trips.

b. This answer is incorrect because the generator load reject is not the most severe transient. This event assumes the reactor scrams on stop valve closure initiation. The candidate who does not realize this may select this answer. This answer is plausible because the event does cause a pressure and power transient.

d. This answer is incorrect because this is not the most severe transient which the SRV safety function is designed to preclude. The stop valve closure will cause a reactor scram and the pressure transient is not the most severe. The candidate who does not recall the bases of this transient may select this answer. This answer is plausible because a pressure perturbation will occur as a result of the transient.

Technical Reference(s):
Procedure 2.1.5, Rev 71 Reactor Scram

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT007-05-04:
2. Discuss the applicable Safety Analysis in the Bases associated with each Section 3.3 Specification.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:
55.41
55.43 6

Difficulty: 3

SRO Only - 10CFR55.43 b (2) Facility operating limitations in the technical specifications and their bases.

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		2
K/A #	295014 G 2.4.23	
Importance Rating		4.4

295014 Inadvertent Reactivity Addition

2.4.23 Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

Question: 84

The plant is operating at 100% power when an ATWS occurs. Reactor Water Level is lowered.

Subsequently the following conditions exist:

- Hot Shutdown Boron Weight injection is complete.
- HPCI is the only available injection system.
- HPCI is restoring RPV level when reactor Power rises to 5% and continues to rise.

What EOP Mitigating Strategy is implemented and why is it chosen?

a. Transition to the main flow path at H to re-perform stop and prevent injection because Cold Shutdown Boron Weight has NOT been injected.

b. Transition to the main flow path at H to re-perform stop and prevent injection because the amount of boron required to shut down the reactor has NOT reached the core.

c. Continue steps in J for injection at a reduced rate because Cold Shutdown Boron Weight has NOT been injected.

d. Continue steps in J for injection at a reduced rate because the amount of boron required to shut down the reactor has NOT reached the core.

Answer:

b. Transition to the main flow path at H to re-perform stop and prevent injection because the amount of boron required to shut down the reactor has NOT reached the core.

Explanation (Optional):

Cooper's PSTG - Step C5-6 directs restoration of RPV water level to the normal range after having injected sufficient boron to shut down the reactor. If reactor power commences and continues to increase as RPV water level is raised, the amount of boron required to shut down the reactor has not reached the core. Returning to Steps C5-3 and C5-4 under these conditions will again require that RPV water level be lowered to prevent flux oscillations and reduce reactor power while additional actions to shut down the reactor proceed.

As injection into the RPV is initially increased to raise RPV water level, a small transient increase in reactor power is expected as natural circulation core flow is re-established. This power increase will be reversed after several seconds as boron is mixed and carried from the lower plenum up into the core region. The wording in the above step, "*commences and continues to increase*," has been specifically chosen to denote only a sustained increase in reactor power indicative of insufficient boron in the core.

Distracters:

a. Stopping and preventing injection helps lower reactor power but Cold Shutdown Boron weight is not the reason this must be done. EOPs wait for Cold Shutdown Boron weight to be injected before reactor depressurization is begun. The candidate that does not recall the bases for cold shutdown boron weight may choose this answer. This answer is plausible because cold shutdown boron weight is an action point in EOPs and stopping and preventing injection is an accurate response for a reactor power rise that is sustained.

c. Reducing the injection rate would be an acceptable response to the reactor power rise but EOPs require stop and prevent because power continues to rise. Injection is not lowered because of Cold shutdown boron weight. EOPs wait for Cold Shutdown Boron weight to be injected before reactor depressurization is begun. The candidate that does not recall the bases for cold shutdown boron weight may choose this answer. This answer is plausible because cold shutdown boron weight is an action point in EOPs and lowering injection of cold water would help lower reactor power.

d. Reducing the injection rate would be an acceptable response to the reactor power rise but not a power rise that continues. EOPs require stop and prevent in this instance. The candidate that does not recall this reasoning may select this answer. This answer is plausible because cold water injection can cause reactor power to rise.

Technical Reference(s):
INT008-06-06, Rev 19 EOP-6A,
INT 008-06-10, Rev 24 EOP-7A, Cooper's PSTG

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT00806100010900, Given an EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM) step, state the reason for the actions contained in the step.
INT00806060011200, Given an EOP flowchart 6A, RPV PRESSURE/POWER step, state the reason for the actions contained in the step.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:

55.41
55.43 5

Difficulty: 4

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation. The SRO is responsible for changing strategies while incorporating EOP actions.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		1
Group #		2
K/A #		295035.AA2.01
Importance Rating		4.1

295035 Secondary Containment High Differential Pressure

EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE:

EA2.01 Secondary containment pressure: Plant Specific

Question: 85

The plant is operating at 100% power when the following conditions occur:

- Un-isolable Steam leak on the HPCI system.
- Reactor Building differential pressure is 0.10 inches of water.
- The maximum area temperatures around the leak are approximately 180ºF steady.
- Reactor Building Exhaust Rad Monitors are reading between 4 to 7 mR/hr and steady.
- Group 6 isolation on low RPV water level.

What strategy is appropriate for the CRS?

a. Enter Procedure 2.1.5, REACTOR SCRAM, to lower radiation being released into the environment.

b. Enter EOP-1A to ensure the plant is scrammed and EOP-2A to Emergency depressurize the vessel in order to restore Secondary Containment temperatures.

c. Enter EOP 5A and defeat isolation interlocks as necessary in order to restart Rx Bldg HVAC and restore Secondary Containment differential pressure.

d. Enter Procedure 5.8.2, RPV DEPRESSURIZATION SYSTEMS, to anticipate Emergency depressurization with the main steam lines in order to lower energy being released into Secondary Containment.

Answer:

c. Enter EOP 5A and defeat isolation interlocks as necessary in order to restart Rx Bldg HVAC and restore Secondary Containment differential pressure.

Explanation (Optional):
Since Secondary Containment differential pressure is less than -0.25 inches of water and the Reactor building ventilation isolated due to a Group 6 isolation caused by RPV level <-42 inches EOP-5A has an override to check the Rx Bldg Exhaust ventilation rad monitors and ensure they

are less than 10 mR/hr and restart normal Rx Bldg ventilation to aid in temperature and pressure control. These instructions are located in an override to the concurrent steps for controlling Secondary Containment Temperature, Radiation and Water Level. Restoring Reactor Building HVAC aids in cooling the building but will restore secondary containment pressure to a negative value.

Distracters:

a. enter Procedure 2.1.5 Reactor Scram to start the cooldown is incorrect because there is a primary system discharging into secondary containment. The radiation levels in secondary containment are elevated but not overly high. A higher priority would be to restore ventilation and preclude a ground level release. A candidate might choose this answer if they misinterpret the radiation levels as being too high. This answer is plausible because the actions stated are appropriate if conditions were of a higher order.

b. enter EOP-1A to ensure the plant is scrammed and EOP-2A to Emergency depressurize the vessel is incorrect because the Area temperatures are not high enough to require these actions. A candidate might choose this answer if they misinterpret the temperatures in the area of the leak and thought they were above the Max Safe levels of 195ºF which require an emergency depressurization. This answer is plausible because the actions stated are appropriate if conditions were worse.

d. anticipate Emergency depressurization with the main steam lines in accordance with Emergency Procedure 5.8.2 RPV DEPRESSURIZATION SYSTEMS is incorrect because the Area temperatures are steady and not currently approaching the Max Safe value. A candidate might choose this answer if they misinterpret the temperatures in the area of the leak and thought they were approaching the Max Safe levels of 195ºF which would allow them to anticipate emergency depressurization. This answer is plausible because anticipating emergency depressurization is a strategy that is implemented if conditions warrant.

Technical Reference(s):
INT008-06-17, Rev 17 EOP 5A

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT00806170010600, Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:
55.41
55.43 5

Difficulty: 4

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		2
Group #		1
K/A #	205000 G 2.1.36	
Importance Rating		4.1

205000 Shutdown Cooling

2.1.36 Knowledge of procedures and limitations involved in core alterations.

Question: 86

A refueling outage is ongoing and a core alteration is being made on 1001'.

What license level is required to fulfill the CHECKER position?
What procedure directs this requirement?

a. An RO is required to fulfill the position.
NPP 10.6, SFSP FUEL STORAGE CONSTRAINTS.

b. An SRO is required to fulfill the position.
NPP 10.6, SFSP FUEL STORAGE CONSTRAINTS.

c. An RO is required to fulfill the position.
NPP 10.21, SPECIAL NUCLEAR MATERIALS CONTROL AND ACCOUNTABILITY INSTRUCTIONS.

d. An SRO is required to fulfill the position.
NPP 10.21, SPECIAL NUCLEAR MATERIALS CONTROL AND ACCOUNTABILITY INSTRUCTIONS.

Answer:

d. An SRO is required to fulfill the position.
NPP 10.21, SPECIAL NUCLEAR MATERIALS CONTROL AND ACCOUNTABILITY INSTRUCTIONS.

Explanation (Optional):
Per NPP 10.21, the SRO is required to be the CHECKER for core alterations. The RO can perform the role of CHECKER for fuel moved within and ICA. Because fuel is being moved from the reactor to the spent fuel pool, the fuel move is between ICAs

Distracters:

a. This answer is incorrect because the wrong license is listed and the incorrect procedure is listed. This answer is plausible because the RO can perform the CHECKER role for some fuel moves and the procedure listed does address positioning of fuel from the reactor to the SFSP. The candidate that does not recall the license level for core alterations may select this answer.

b. This answer is incorrect because the incorrect procedure is listed. This answer is plausible because the SRO must perform the CHECKER role for this fuel move (core alteration) and the procedure listed does address positioning of fuel from the reactor to the SFSP. The candidate that does not recall the correct procedure for core alterations may select this answer.

c. This answer is incorrect because the wrong license is listed. This answer is plausible because the RO can perform the CHECKER role for some fuel moves and the correct procedure is listed. The candidate that does not recall the license level for core alterations may select this answer.

Technical Reference(s):
NPP 10.21, Rev 50, SPECIAL NUCLEAR MATERIALS CONTROL AND ACCOUNTABILITY INSTRUCTIONS

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT0070501001030A From memory, define the following terms: Core Alteration

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:
55.41
55.43 6

Difficulty: 3

SRO Only - 10CFR55.43 b (6) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity. The SRO is responsible for supervising core alterations).

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		2
Group #		1
K/A #		211000.A2.06
Importance Rating		3.3

211000 SLC

A.2 Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.06 Valve Openings

Question: 87

A clearance order error has resulted in the SLC tank level lowering from 80% to 58% on SLC-LI-66 (Panel 9-5). An ATWS subsequently occurs requiring SLC injection with the tank level at 58%.

How is the SLC system affected?
What procedure provides guidance for the SLC system recovery?

a. Hot Shutdown Boron weight conditions cannot be met if required.
Procedure 2.2.74, STANDBY LIQUID CONTROL SYSTEM provides tank restoration guidance.

b. Cold Shutdown Boron weight conditions cannot be met if required.
Procedure 2.2.74, STANDBY LIQUID CONTROL SYSTEM provides tank restoration guidance.

c. Hot Shutdown Boron weight conditions cannot be met if required.
Procedure 5.8.8, ALTERNATE BORON INJECTION AND PREPARATION provides tank restoration guidance.

d. Cold Shutdown Boron weight conditions cannot be met if required.
Procedure 5.8.8, ALTERNATE BORON INJECTION AND PREPARATION provides tank restoration guidance.

Answer:

d. Cold Shutdown Boron weight conditions cannot be met if required.
Procedure 5.8.8, ALTERNATE BORON INJECTION AND PREPARATION provides restoration guidance.

Explanation (Optional):

Cold shutdown boron weight is the injection of 60% of the SLC tank contents. Hot shutdown boron weight is the injection of 26% of the SLC tank contents. With the SLC tank level at 58%, cold shutdown boron weight cannot be reached. Hot shutdown boron weight can be accomplished. In order to reach cold shutdown boron weight, additional demineralized water and borax and boric acid must be added to the tank. The procedure guidance for emergency boron preparation is procedure 5.8.8. Section 8 provides the direction for filling the tank to 52% and the amount of chemicals to add to come up with a 15% sodium pentaborate solution by weight. The requirement to air sparge the tank contents for 4 hours is not required per this procedure. Procedure 2.2.74 provides guidance for adding chemicals to the SLC storage tank but only after sparging for 4 hours and sampling by chemistry. This procedure is not the correct procedure to use during an emergency condition (ATWS).

Distracter:

a. is incorrect because hot shutdown boron weight conditions can be met. The procedure listed is <u>not</u> the correct procedure for emergency conditions. This answer is plausible because the procedure listed does provide guidance for mixing the tank solution. The candidate that cannot recall hot shutdown boron weight or the correct procedure may select this answer.

b. is incorrect because the listed procedure is <u>not</u> the correct procedure to use during emergency conditions. This answer is plausible because cold shutdown boron weight cannot be injected without additional inventory being mixed. The candidate that cannot recall the correct procedure may select this answer.

c. is incorrect because hot shutdown boron weight conditions can be met. The procedure listed is the correct procedure for emergency conditions. This answer is plausible because the procedure listed does provide guidance for mixing the tank solution. The candidate that cannot recall hot shutdown boron weight or the correct procedure may select this answer.

Technical Reference(s):
EOP 5.8.8, Rev 14 Alternate Boron Injection and Preparation

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR00229020011100 Given a specific SLC malfunction determine the effect it would have on the ability to shutdown the reactor.

Question Source:

Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content
55.41
55.43 5
:

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		2
Group #		1
K/A #	215005	G 2.2.22
Importance Rating		4.7

215005 APRM/LPRM

2.2.22 Knowledge of limiting conditions for operations and safety limits.

Question: 88

The plant is 100% power with the APRM Rod Block testing due today. 6.1APRM.305, APRM SYSTEM (FLOW BIAS AND STARTUP) CHANNEL CALIBRATION (DIV 1), is to be performed.

- Yesterday, APRM A was declared inoperable due to too few LPRM inputs and will NOT be tested.
- APRM A remains un-bypassed.
- The test is started and at 0800 APRM C and APRM F are bypassed as required.
- At 0815 with APRM C and F still in BYPASS:
  - I&C reports the downscale trip for APRM C is at 2%.
  - I&C reports that the downscale trip can be adjusted.

What is(are) the required action(s) to maintain or re-establish compliance with Tech Specs or the TRM?

a. Declare APRM C inoperable and insert a rod block by 0915 because one required channel is inoperable.

b. Immediately enter LCO 3.0.3 and start lowering power by 0915 because two required channels are inoperable.

c. Satisfactorily complete APRM C testing by 1415 or insert a rod block by 1515 since the time that a channel can be bypassed without entering the required action expires.

d. Satisfactorily complete ARPM C testing by 1400 or enter LCO 3.0.3 and lower power by 1500 since the time that a channel can be bypassed without entering the actions expires.

Answer:

a. Declare APRM C inoperable and insert a rod block by 0915 because one required channel is inoperable.

Explanation (Optional):

There is a note modifying the TS and TSR requirements that allows a channel to be bypassed for testing without entering the actions although the channel is inoperable provided the associated function maintains RPS (TS) or rod block (TRM) capability. The Delayed Entry Time is 6 hours. However, per TSR 3.0.1, failure to meet a surveillance, whether such failure is experienced during or between performances of the surveillance, shall be failure to meet the LCO or TLCO. A note allows a separate Condition entry for each channel. The scram function for the LCO requires 2 channels per trip system. The Control Rod Block function for the TLCO requires 4 channels. There are 6 APRMs. Three APRMS per RPS trip system. Four APRM required channels for enforcing a control rod blocks.
APRM A is inoperable due to not enough LPRM inputs (INOP). APRMs C and F are inoperable for testing but the condition and required action entry can be delayed for 6 hours for both TS and the TRM. As long as APRM C is repaired by 1600 the condition and required actions of TS or the TRM do not have to be entered.

Even though the number of channels does not meet the TS required number, there is at least one APRM available to cause a rod block, so the rod block function is maintained. There is one APRM in each RPS channel that can cause an RPS trip so the scram function is maintained. Because both the rod block and the scram functions are maintained, the DET does not apply.

Distracters**:**

b. is incorrect. There are is only one APRM that is inoperable requiring conditions and required actions to be entered. APRMs C and F are inoperable for testing but the LCO and TLCO continues to be met because of the allowable DET time of 6 hours. The candidate that does not understand delayed entry times may choose this answer. This answer is plausible because 3 channels are inoperable.

c. is incorrect. SR 3.0.1 requires declaring the APRM inoperable and entering the actions by the DET time of 1400 (the time the APRMS were first bypassed). At 1400 the DET expires requiring the rod block to be inserted if APRM C is not repaired and operable. The candidate that does not understand delayed entry times may choose this answer. This answer is plausible because the DET time if added to the time APRM C downscale is out of tolerance is 1415.

d. is incorrect. SR 3.0.1 requires declaring the APRM inoperable and entering the actions by the DET time of 1400 (the time the APRMS were first bypassed). LCO 3.0.3 is not required to be entered at this time because the required action would be to place one channel in the trip condition in 6 hours. After the 6 hour time limit is exceeded a shutdown would be required. The candidate that does not understand delayed entry times may choose this answer. This answer is plausible because LCO 3.0.3 does exist for LCO completions times that cannot be met.

Technical Reference(s):
TRM 3.3.1, TS 3.3.1.1

Proposed references to be provided to applicants during examination:
TRM 3.3.1, Control Rod Block Instrumentation
TS LCO 3.3.1, Reactor Protection System (RPS) Instrumentation
TS LCO 3.3.3.1, PAM Instrumentation

Learning Objective:

Given plant conditions and the TRM, determine the ACTIONS required per the TLCOs:
T.3.3.1 Control Rod Block Instrumentation

Question Source:

Bank # 11504
Modified Bank #
New

Question Cognitive Level:

Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:

55.41
55.43 2

Difficulty: 4

SRO Only - 10CFR55.43 b (2) Facility operating limitations in the technical specifications and their bases. The SRO is responsible for TS and TRM instrumentation operability and surveillance requirements.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		2
Group #		1
K/A #	262001.A2.07	
Importance Rating		3.2

262001 AC Electrical Distribution

A2. 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:

A2.07 Energizing a dead bus

Question: 89

The CRS has entered EOP 2A and steam cooling is being utilized to cool the core. The RPV pressure is steady at 250 psig using SRVs. The CRS is also in 5.3SBO, STATION BLACKOUT. The dispatcher has called and reported the Emergency transformer is ready to be energized.

When the critical 4160V buses are re-energized, what system(s) inject?
With RPV water level at -190 inches and rising, what action is required next?

a. All low pressure ECCS systems start and inject into the RPV.
Transition to EOP 1A to control and restore RPV level.

b. All low pressure ECCS systems start and inject into the RPV.
Continue actions in EOP 2A and emergency depressurize the RPV.

c. All low pressure ECCS systems start but only CS injects into the RPV.
Continue actions in EOP 2A and emergency depressurize the RPV.

d. All low pressure ECCS systems start but only CS injects into the RPV.
Transition to EOP 1A to control and restore RPV level.

Answer:

a. All low pressure ECCS systems start and inject into the RPV.
Transition to EOP 1A to control and restore RPV level.

Explanation (Optional):

The CRS is in steam cooling and EOP 2A only when no source of injection is available to the RPV. With RPV pressure at 250 psig, Core Spray and RHR can inject into the RPV. When the Emergency Transformer is re-energized, supply breaker is closed per 5.3SBO and the RHR and CS pumps start on low level/drywell pressure start signals. The RHR system injects at RPV pressures below 275 and the Core Spray system injects at RPV pressures below 350 psig. RPV level immediately starts rising when both Core Spray and 4 RHR pumps start injecting. Once RPV level starts rising, EOP 2A override directs the CRS to transition to EOP 1A for RPV level control.

Distracters:

b. This answer is incorrect because emergency depressurization is not the correct strategy when RPV level is rising. All low pressure ECCS do start and inject at the given RPV pressure. The override in the steam cooling leg of 2A directs transitioning to EOP 1A. This answer is plausible because there is an override in 2A to emergency depressurize with a source injecting but only if RPV level cannot be restored above -183 inches. The candidate who does not recall the override to transition to EOP 1A would select this answer.

c. This answer is incorrect because RHR also injects at the given RPV pressure and emergency depressurization is not the correct strategy when RPV level is rising. The override in the steam cooling leg of 2A directs transitioning to EOP 1A. This answer is plausible because there is an override in 2A to emergency depressurize with a source injecting but only if RPV level cannot be restored above -183 inches. The candidate who does not recall the override to transition to EOP 1A would select this answer.

d. This answer is incorrect because RHR also injects at the given RPV pressure. This answer is plausible because the strategy to transition to EOP 1A is correct. The candidate who does not recall the RPV pressure at which RHR injects would select this answer.

Technical Reference(s):
Student Text, COR0010102, Rev 17 AC Electrical Distribution
Student Text INT0080607, Rev 19, EOP 2A Emergency Depressurization & Steam Cooling

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR0010102001080H, Predict the consequences of the following on plant operation: Loss of Off-site Power
COR0010102001130G, Predict the consequences of the following events on the AC Electrical Distribution System: Energizing a dead bus
INT0080607 2. Describe the RPV conditions that require steam cooling and the reason for steam cooling.

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

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

10 CFR Part 55 Content:
55.41
55.43 5

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation. It is the SROs responsibility to change plant priorities and execute the appropriate procedures for the priority change.

ES-401 **Written Examination Question Worksheet** **Form ES-401**

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		2
Group #		1
K/A #	264000 G 2.4.28	
Importance Rating		4.1

264000 EDGs

2.4.28 Knowledge of procedures relating to a security event (non-safeguards information).

Question: 90

The plant is operating at 100% power when the Shift Manager receives a "credible imminent threat" notification from the NRC.

Concerning the Emergency Generators what actions are required?

Enter Procedure…

a. 5.5Security and start and load both Emergency Diesel Generators.

b. 5.5Security and start both Emergency Diesel Generators and run unloaded.

c. 5.3ALT-Strategy and start and load both Emergency Diesel Generators.

d. 5.3ALT-Strategy and start both Emergency Diesel Generators and run unloaded.

Answer:

b. 5.5Security and start both Emergency Diesel Generators and run unloaded.

Explanation (Optional):

When the NRC calls the Shift Manager and notifies him of an imminent <u>or</u> actual threat <u>specific to CNS</u>. The Shift Manager will enter Emergency Procedure 5.5SECURITY and proceed to Attachment 3 for a Credible, Imminent or Actual Threat. In that attachment are steps to perform Attachment 9 Threat Driven Plant Actions one of which is to start and run unloaded both Emergency Diesel Generators.

Distracters**:**

a. 5.5Security and start and load both emergency diesel generators, this is incorrect because 5.3ALT-Strategy only provides guidance for performing an emergency diesel generator black start. A candidate might choose this answer if they remembered the correct procedure but failed to apply the logic correctly to the DGs and forgot that they would strip the bus and retie on a loss of power.

c. 5.3ALT-stragety and start and load both emergency diesel generators, this is incorrect because it is the wrong procedure and the EDGs are not to be loaded. A candidate might choose this answer if they confused the guidance of starting the DGs and mistakenly placed it in 5.3ALT-Strategy.

d. 5.3ALT-stragety and start both emergency diesel generators and run unloaded, this is incorrect because 5.3ALT-Strategy only provides guidance for performing an emergency diesel generator black start. A candidate might choose this answer if they confused the guidance of starting the DGs and mistakenly placed it in 5.3ALT-Strategy.

Technical Reference(s):
5.3ALT-STRATEGY, Rev 42, Alt Core Cooling Mitigating Strategies
5.5SECURITY Rev 38, Security

Proposed references to be provided to applicants during examination: None

Learning Objective:
Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

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

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

10 CFR Part 55 Content:
55.41
55.43 5

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation. It is the SROs responsibility to select the appropriate procedure execute the attachments for this event.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		2
Group #		2
K/A #	201003.A2.10	
Importance Rating		3.4

201003 Control Rod and Drive Mechanism

A2. Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.10 Excessive SCRAM time for a given drive mechanism

Question: 91

The control rods listed below have been scram time tested at 1000 psig reactor pressure, with the indicated results.

Control Rod	Time	Control Rod	Time
30-19	7.24 sec. to notch 06	22-51	2.07 sec. to notch 26
06-43	2.98 sec. to notch 06	38-35	1.39 sec. to notch 26
38-47	3.16 sec. to notch 06	14-19	1.18 sec. to notch 36
10-47	3.31 sec. to notch 06	14-27	0.61 sec. to notch 36
18-47	2.91 sec. to notch 06	42-43	0.97 sec. to notch 36
22-11	2.89 sec. to notch 06	46-19	0.34 sec. to notch 46
42-15	2.96 sec. to notch 06	46-43	0.44 sec. to notch 46
50-19	3.27 sec. to notch 06	06-39	0.51 sec. to notch 46

How many of the rods tested must be classified as "slow" control rods?

a. 2

b. 3

c. 4

d. 5

Answer:

b. 3

Explanation (Optional):
Controls rods 06-39, 22-51, and 46-19 are classified as "slow". Rod 30-19 must be declared inoperable based on NOTE 2 for Table 3.1.4-1.

Distracters:

a. More than 2 rods exceed Tech Spec limits. This answer is plausible because two control rods are slow. The candidate that does not realize another control rod is inoperable may select this answer.

c. Only 3 control rods are slow. Rod 30-19 must be declared inoperable and is not considered "slow" per NOTE 2 for Table 3.1.4-1. This answer is plausible because 4 control rods are inserting slowly. The candidate that does not realize the NOTE applies would select this answer.

d. Only 3 control rods are slow and one control rod is so slow it must be declared inoperable. This answer is plausible because some control rods are close to being slow. The candidate that does not correctly read table 3.4.1-1 may select this answer.

Technical Reference(s):
T.S. 3.1.4 Control rod scram times

Proposed references to be provided to applicants during examination: LCO 3.1.4, and Table 3.4.1-1, Figure of core map

Learning Objective:

INT00705020010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.1 LCO, determine the ACTIONS that are required.
INT00705020010100	Given a set of plant conditions, recognize non-compliance with a Section 3.1 LCO.

Question Source:
Bank # 9256
Modified Bank #
New

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

10 CFR Part 55 Content:
55.41
55.43 2

Difficulty: 3

SRO Only - 10CFR55.43 b (2) Facility operating limitations in the technical specifications and their bases.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		2
Group #		2
K/A #	230000.A2.07	
Importance Rating		3.8

230000 RHR/LPCI: Torus/Pool Spray Mode

A2. Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.07 Emergency generator failure

Question: 92

The plant is operating at 100% power with the Emergency Transformer out of service for maintenance when the following occurs:

- Startup Transformer locks out.
- A reactor steam leak in the drywell results in a torus pressure of 11 psig (slowly rising)
- Reactor Water Level is being maintained by the HPCI system between -20 to +30 inches on the Wide Range instruments

The panel operator places Loop "A" RHR in Torus and Drywell Sprays, with both RHR Pumps running as directed. Drywell pressure starts to lower very slowly. Two minutes later the following occurs.

- Diesel 1 trips and locks out.

Are Torus and Drywell sprays still in service?
What actions mitigate the effects of the loss of the Diesel Generator?

a. **No.**
Direct re-establishing sprays per Procedure 2.2.69, RESIDUAL HEAT REMOVAL SYSTEM.

b. **No.**
Direct re-establishing sprays per Procedure 2.2.69.3, RHR SUPPRESSION POOL COOLING AND CONTAINMENT SPRAY.

c. **Yes**.
Direct establishing additional spray flow per Procedure 2.2.69, RESIDUAL HEAT REMOVAL SYSTEM.

d. **Yes**.

Direct establishing additional spray flow per Procedure 2.2.69.3, RHR SUPPRESSION POOL COOLING AND CONTAINMENT SPRAY.

Answer:

d. **Yes**.
Direct establishing additional spray flow per Procedure 2.2.69.3, RHR SUPPRESSION POOL COOLING AND CONTAINMENT SPRAY.

Explanation (Optional):

With the Emergency Transformer out of service for maintenance and the lockout of the Startup Transformer, the Diesels to start and load the 1F and 1G Critical 4160 VAC Busses. The Panel Operator starts both the 1A and 1C RHR Pumps, 1A is powered from the 1F 4160 VAC Bus which is being supplied by Diesel 1. 1C RHR Pump is powered from the 1G 4160 VAC Bus and Diesel 2.

The Torus cooling, Torus and Drywell Spray valves are powered by Div I AC which is getting its power from Diesel 1, so they fail as is.

Torus Cooling, Torus and Drywell sprays continue at a reduced rate due to the tripping of the 1A RHR Pump when the Diesel trips. Since the Drywell pressure was just very slowly lowering with two RHR Pumps in service, the reduction in flow causes a reduction in containment cooling. The other loop of RHR must be placed in service, the Div II powered pump in that loop restores flow to previous rates. Since there is another heat exchanger that will be used, the rate of DW Pressure drop should increase.

Distracters:

a. **No** and sprays should be re-established per Procedure 2.2.69. This is incorrect because one RHR pump powered from Diesel 2 will continue to run in loop A RHR spray. This is also the wrong procedure. A candidate may choose this answer if they forget that the pumps are not powered from the same Division Diesel Generator. This answer is plausible because the divisional power supply to the RHR pumps is not division specific for two of the RHR pumps and the procedure provides guidance for RHR system operation other than spray cooling mode.

b. **No** and sprays should be re-established per Procedure 2.2.69.3. This is incorrect because one RHR pump powered from Diesel 2 will continue to run in loop A RHR spray. A candidate may choose this answer if they forget that the pumps are not powered from the same Division Diesel Generator. This answer is plausible because the divisional power supply to the RHR pumps is not division specific for two of the RHR pumps.

c. **Yes.** Although establishing additional sprays is the correct action, the procedure is incorrect. A candidate may choose this answer if they forget the correct procedure. This answer is plausible because the procedure provides guidance for RHR system operation other than spray cooling mode.

Technical Reference(s):
Procedure 2.2.69, Rev 97 Residual Heat Removal System

Procedure 2.2.69.3, Rev 46 RHR Suppression Pool Cooling And Cont Spray

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022302001030P, Describe RHR System design feature(s) and/or interlocks which provide for the following: Spray flow cooling.
COR0022302001080A,Predict the consequences a malfunction of the following will have on the RHR system: A.C. electrical power (including RPS)
COR0022302001170B,Given plant conditions, determine actions necessary to place RHR in the following flowpaths: Suppression Chamber Spray
COR0022302001170C, Given plant conditions, determine actions necessary to place RHR in the following flowpaths: Drywell Spray

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level:

Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:

55.41
55.43 5

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation. It is the SROs responsibility to select the appropriate procedure for a given event.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		2
Group #		2
K/A #	268000 G	2.2.42
Importance Rating		4.6

268000 Radwaste

2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question: 93

The plant is operating at 100% power during a liquid release of Waste Sample Tank A that has been going for about 3 hours at a flow rate of 30 gpm, when you take the watch. Today is May 24, 2014.

While reviewing the release authorization, you discover that the concentration of radioactive materials in the tank were 1.0 E-1 µci/ml (excluding tritium and noble gases) before treatment as well as following treatment.

What action(s) is/are **required**?

Prepare and submit a Special Report to the NRC by...

a. June 24, 2014, identifying equipment not OPERABLE, actions taken to restore the inoperable equipment to OPERABLE status and a summary description of the actions taken to prevent recurrence.

b. June 24, 2014, that 1) defines actions to be taken to reduce releases and prevent recurrence, and 2) results of an exposure analysis to determine whether the dose or dose commitment to a Member of the Public.

c. July 31, 2014, identifying equipment not OPERABLE, actions taken to restore the inoperable equipment to OPERABLE status and a summary description of the actions taken to prevent recurrence.

d. July 31, 2014, that 1) defines actions to be taken to reduce releases and prevent recurrence, and 2) results of an exposure analysis to determine whether the dose or dose commitment to a Member of the Public.

Answer:

c. July 31, 2014, identifying equipment not OPERABLE, actions taken to restore the inoperable equipment to OPERABLE status and a summary description of the actions taken to prevent recurrence.

Explanation (Optional):

In accordance with DLCO 3.1.2, Condition A has been exceeded, after the liquid waste discharge. Requiring condition B.1 to be entered immediately and Required action B.1 “Prepare and submit a Special Report to the NRC, identifying equipment not OPERABLE, actions taken to restore the inoperable equipment to OPERABLE status and a summary description of the actions taken to prevent recurrence" with a completion time of 31 days following the end of the quarter in which the limit was exceeded. Since the limit was exceeded on May 24th, the end of that quarter would then be June 30th. 31 days added to that is July 31st.

Distracters:

a. This answer is 31 days after the occurrence, not 31 days following the end of the quarter. This answer is plausible because of the 31 day reference and when it is applied. The candidate that does not comprehend the quarter statement may choose this answer.

b. This answer is 31 days after the occurrence, not 31 days following the end of the quarter. Plus this is the action for DLCO 3.1.3 Liquid Effluents Dose, not DLCO 3.1.2 Liquid Waste Concentrations. This answer is plausible because of the 31 day reference and when it is applied. The candidate that does not comprehend the quarter statement may choose this answer.

d. This answer is the action for DLCO 3.1.3 Liquid Effluents Dose, not DLCO 3.1.2 Liquid Waste Concentrations; however it is done by the correct date. This candidate that mis-interprets the statement of the DLCO may select this answer. This answer is plausible because DLCO 3.1.2 does exist for concentrations.

Technical Reference(s):
DLCO 3.1.3 Liquid Effluents Dose
DLCO 3.1.2 Liquid Waste Concentrations

Proposed references to be provided to applicants during examination:
DLCO 3.1.3 Liquid Effluents Dose,
DLCO 3.1.2 Liquid Waste Concentrations

Learning Objective:
Given the CNS ODAM Appendix D and a set of conditions, determine any special reporting requirements.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:

55.41
55.43 2

Difficulty 3

SRO Only - 10CFR55.43 b (2) Facility operating limitations in the technical specifications and their bases. The SRO is responsible for release rates being within limitations.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		3
Group #		1
K/A #	2.1.7	
Importance Rating		4.7

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Question: 94

Reactor power is being raised to rated after a down power to perform a rod sequence exchange. As RR speed is raised to 97% reactor power, the following is noted:

- PMIS Point B021, REACTOR WATER LEVEL A, has cycled in a 6.0 inch level band (peak-to-peak) 7 times in the last 4 minutes.
- PMIS Point N011, REACTOR WATER LEVEL B, has cycled in a 1.1 inch level band (peak-to-peak) 6 times in the last 4 minutes.

What action is appropriate to direct?

a. Per Procedure 2.1.10, STATION POWER CHANGES, lower RR speed until the level cycles on both PMIS points are at or below 5.5 inches peak to peak.

b. Per Procedure 2.0.3.1, OPERATIONAL STRATEGY GUIDANCE PROCESS, hold RR speed steady until the reason for the anomaly can be determined.

c. Per Procedure 2.4RXLVL, RPV WATER LEVEL CONTROL TROUBLE, stop raising RR pump speed and bypass NBI-LT-52A and NBI-LT-52C on the RVLC MAINT screen.

d. Per Procedure 2.4RR, REACTOR RECIRCULATION ABNORMAL, lower RR pump speed until all PMIS points are steady for a minimum of 5 minutes.

Answer:

a. Per Procedure 2.1.10, STATION POWER CHANGES, lower RR speed until the level cycles on both PMIS points are at or below 5.5 inches peak to peak.

Explanation (Optional):

Procedure 2.1.1, Section 4, RPV LEVEL OSCILLATION GUIDELINES, directs lowering power with RR pumps until sustained level oscillations are at or below 5.5 inches peak-to-peak. The procedure guidance is deep in the procedure that is not referenced by the Precautions and Limitations section. The information is not a NOTE or CAUTION either. The SRO must be familiar with the oscillation phenomenon and which procedure contains the strategy to mitigate its effects.

Distracters:

b. Procedure 2.0.3.1 is used when no other procedure guidance is known and a condition exists. In this instance, Procedure 2.1.10 has guidance to lower reactor power with RR pumps until the oscillations are at or below 5.5 inches peak-to-peak. A candidate that does not recall the procedure or the required guidance (lower RR speed) may select this answer. This answer is plausible because the process in Procedure 2.0.3.1 exists to put a plan in place if a condition is not understood or current procedure guidance does not exist.

c. Procedure 2.4RXLVL does provide guidance to bypass instruments only if A and C instruments are not trending with B and D instruments. In this instance the D instrument trend is not known and there is no reason to assume it would be trending any differently than instrument B. The instruments are oscillating anyway. The candidate that does not recall the guidance in Procedure 2.1.10 may select this answer. This answer is plausible because it is actions that would be taken for RPV level mismatches.

d. There is no procedure guidance to monitor level for 5 minutes. Procedure 2.4RR does not require action to lower power until oscillations are < 5.5 inches. The candidate that does not recall the guidance in Procedure 2.1.10 may select this answer. This answer is plausible because a lot of times the operator will stop the current action (raising power) and monitor the plant until the anomaly is known.

Technical Reference(s):
2.1.10, Rev 106 Station Power Changes
2.0.3, Rev 86 Conduct Of Operations

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT032010400F030B Discuss the following as described in Procedure 2.1.10, Station Power Changes: Raising Power

Question Source:
Bank # 25377
Modified Bank #
New

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

10 CFR Part 55 Content:
55.41
55.43 5

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situation. It is the SROs responsibility to change plant priorities and execute appropriate counter-measures for unexpected plant conditions.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		3
Group #		1
K/A #	2.1.35	
Importance Rating		3.9

G2.1.35: Knowledge of the fuel-handling responsibilities of SROs,

Question: 95

The plant is restoring from RPV refueling. The removal of the Reactor Cavity Work Platform (in Vessel Inspection Platform) is being performed by maintenance.

What is the Refueling Floor SRO's responsibility while this activity is being performed?

a. Monitoring the reactor cavity water level.

b. Ensuring the skimmer surge tank is low.

c. Monitoring the fuel pool for overflow into the exhaust ducting.

d. Ensuring the overhead crane remains in the "restricted zone."

Answer:

a. Monitoring the reactor cavity water level.

Explanation (Optional):

From Procedure 2.1.20.1 Page 3.

4.7 Remove Reactor Cavity Work Platform (in Vessel Inspection Platform) while maintaining cavity and Skimmer Surge Tank levels as follows:

4.7.1 Ensure cavity water level monitored by Refuel Floor SRO or designee.

Distracters:

b. This is incorrect because the reactor well/fuel pool water level will drop when the platform is removed from the water. There will be no overflow into the skimmer surge tank. This answer is plausible because there is a requirement to ensure the skimmer surge tank is near its lower level mark prior to lowering the platform into the reactor well.

c. This is incorrect because the reactor well/fuel pool water level will drop when the platform is removed from the water. The water level rises when the platform is lowered into the reactor well. The candidate that confused this may select this answer. This answer is plausible because past CNS OE covers overflowing the fuel pool while lowering a spent fuel cask into the fuel pool.

d. This is incorrect because the restricted zone reactor building crane operation is only utilized during spent fuel cask transfer. The candidate that is not fully aware of when the reactor building crane must be operated in the restricted zone may select this answer. This answer is plausible because the restricted mode operation for the reactor building crane is required for fuel cask handling.

Technical Reference(s):
2.1.20.1, Rev 31 Restoration From RPV Refueling

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT0320139B0B0100, Describe the general sequence of events performed during restoration form RPV refueling per Procedure 2.1.20.1, Restoration From RPV Refueling.

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content:
55.41
55.43 7

Difficulty 2

SRO Only - 10CFR55.43 b (7) Fuel handling facilities and procedures. It is the SROs responsibility as refueling floor supervisor for supervising fuel handling activities and the procedures that govern the activities.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		3
Group #		2
K/A #	2.2.5	
Importance Rating		3.2

2.2.5: Knowledge of the process for making design or operating changes to the facility.

Question: 96

What condition for a proposed plant modification must have prior approval from the NRC?

The modification requires a...

a. 50.59 evaluation.

b. revision to the UFSAR.

c. change to the Emergency Plan.

d. Technical Specification amendment.

Answer:

d. Technical Specification amendment.

Explanation (Optional):

Technical Specification amendments cannot be made without approval from the NRC first. The 50.59 evaluation is used to determine if prior NRC approval is required but is not a given. The revisions to USAR are reviewed by the NRC but are reviewed periodically after the fact. The E Plan is controlled by 50.54q evaluations that may or may not require prior approval.

Distracters:

a. incorrect, this evaluation will determine if NRC approval is required. The candidate may select this answer if they aren't familiar with the 50.59 process. This answer is plausible because the 50.59 process is utilized to determine if prior NRC approval is required prior to modification implementation.

b. incorrect, must have a 50.59 evaluation but not necessarily NRC approval. The USFAR can be revised without prior NRC approval but the NRC does review all USFAR revision post-implementation. The candidate may select this answer if they do not understand the USFAR revision process. This answer is plausible because the USFAR is a design basis document that requires the NRC review post-revision.

c. incorrect, must have a 50.54(q) evaluation but not necessarily NRC approval. The candidate that is not familiar with the Emergency Plan change process may select this answer. Changes can be made to the E Plan without prior NRC approval but the 50.54(q) process must determine this.

Technical Reference(s):
0-EN-LI-100, Rev 12C0 “Process Applicability Determination”

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT032010100M010A, Discuss the following as described in Administrative Procedure 0-EN-LI-100, “Process Applicability Determination”
INT032010100M010A, The process for the performance and documentation of evaluations pursuant to 10CFR50.59 AND 10CFR72.48

Question Source:
Bank # Perry 2009
Modified Bank # (Note changes or attach parent)
New

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

10 CFR Part 55 Content:
55.41
55.43 3

History: 2009 Perry NRC Exam - SRO Question 21 (See attached)

Difficulty 2

SRO Only - 10CFR55.43 b (3) Facility licensee procedures required to obtain authority for design and operating changes in the facility. The SRO is responsible for ensuring changes to the facility are performed according to the 50.59 process.

Perry NRC EXAM - 2009

QUESTION SRO 21

A proposed plant modification must always have prior approval from the NRC if ____.

A. it requires a 50.59 evaluation

B. it involves a system described in the UFSAR

C. it involves a system included in the Technical Specifications

D. it results in a design basis limit for Primary Containment being altered

K/A# 2.2.5 K&A: Knowledge of the process for making design or operating changes to the facility. 3.2
Equipment Control

Explanation: Answer D – design bases alteration requires a license amendment prior to implementation.

A – incorrect, this evaluation will determine if NRC approval is required

B - incorrect, must have a 50.59 evaluation but not necessarily NRC approval

C - incorrect, must have a 50.59 evaluation but not necessarily NRC approval

Technical Reference(s): 10-CFR-50.59
NOP-LP-4003 Rev 4, NOP-CC-2003 Rev 14
Reference Attached: Forms NOP-LP-4003-01, 02 and 03

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3039 Admin and OT-3037-01

Question Source: Bank # River Bend 2004

Modified Bank #
New
Question History: Previous NRC Exam River Bend 2004
Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis
10 CFR Part 55 Content: 55.41
55.43 X b.3
Comments: Level of Difficulty = 3

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		3
Group #		2
K/A #	2.2.17	
Importance Rating		3.8

2.2.17: Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

Question: 97

The plant is operating at power and an upcoming work week schedule is being prepared.

When is the on-line work schedule frozen?
What procedure provides the guidance?

a. T- 5 week
0-CNS-WM-101, WORK WEEK PROCESS

b. T- 6 week
0-CNS-WM-105, PLANNING

c. T- 5 week
0-CNS-WM-105, PLANNING

d. T- 6 week
0-CNS-WM-101, WORK WEEK PROCESS

Answer:

a. T- 5 week
0-CNS-WM-101, WORK WEEK PROCESS

Explanation (Optional):

Procedure 0-CNS-WM-101, WORK WEEK PROCESS, Section 16 states the schedule is frozen when approved by the Work Week Director at the T-5 meeting.

Distracters:

b. This is incorrect because the schedule is not frozen at T-6. The procedure is incorrect as the planning procedure does not freeze the schedule. A candidate that is not familiar with the work control process may select this answer. This answer is plausible because the work scope is frozen at T-16.

c. This answer is incorrect because Procedure 0-CNS-WM-105, PLANNING does not freeze the work schedule. A candidate that is not familiar with the work control process may select this answer. This answer is plausible because the work schedule is frozen at T-5.

d. This answer is incorrect because the schedule is not frozen at T-6. Procedure 0-CNS-WM-101, WORK WEEK PROCESS does freeze the work schedule but it is frozen at T-5. A candidate that is not familiar with the work control process may select this answer. This answer is plausible because the correct procedure is given.

Technical Reference(s):
0-CNS-WM-101,Rev 6 WORK WEEK PROCESS

Proposed references to be provided to applicants during examination: None

Learning Objective:

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

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

10 CFR Part 55 Content:
55.41
55.43 5

Difficulty 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The SRO is responsible for being familiar with the work week schedule.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		3
Group #		3
K/A #	2.3.6	
Importance Rating		3.8

2.3.6: Ability to approve release permits.

Question: 98

Discharge of the contents of the Floor Drain Sample Tank (FDST) to the river is complete.

At the conclusion of the release of the FDST, what does the duty Shift Manager's signature on 8.8.11 Att. 1 (LIQUID RADIOACTIVE WASTE DISCHARGE FORM) Section 5 (To: Chemist-From Shift Manager) confirm?

a. **Only** that the discharge valve is closed and all the data in Attachment 1 Sections 4 and 5 is correct.

b. Discharge valve is closed, rad monitor is flushed and all data in Attachment 1 Sections 4 and 5 is correct.

c. Release flow rate is 0 gpm, and the total volume released is equal to or less than volume estimated on Attachment 1, Section 1.

d. Release flow rate is 0 gpm, and discharge radiation monitor alarm set point specified in Section 3 was not exceeded during the discharge.

Answer:

b. Discharge valve is closed, rad monitor is flushed and all data in Attachment 1 Sections 4 and 5 is correct.

Explanation (Optional):

Procedure 8.8.11 Step 4.30 states that "Upon completion of discharge, Shift Manager's signature is confirmation discharge valve is closed, monitor has been flushed, and all data in Attachment 1, Sections and, is correct."

Distracters:

a. is incorrect because the signature also indicates also that the monitor has been flushed, and all data in Attachment 1, Sections 4 and 5, is correct. This answer is plausible because part of the listed criteria is correct. The candidate who does not recall the requirement to have the rad monitor flushed may choose this answer.

c. is incorrect although the first condition would certainly exist but the requirement is a closed discharge valve. This answer is plausible because the release is below the limit

calculated. The candidate who cannot recall all the criteria the signature is represents may choose this answer.

d. is incorrect although the first condition would certainly exist the requirement is a closed discharge valve. This answer is plausible because the release is below the limit calculated and the discharge has stopped. The candidate who cannot recall all the criteria the signature is represents may choose this answer.

Technical Reference(s):
8.8.11, Rev 31 Liquid Radioactive Waste Discharge Authorization

Proposed references to be provided to applicants during examination:
Procedure 8.8.11 Attachment 1 only

Learning Objective:
State who, by title, authorizes releases of radioactive liquid effluents from CNS.

Question Source:
Bank # 20525
Modified Bank #
New

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

10 CFR Part 55 Content:
55.41
55.43 4

Difficulty 2

SRO Only - 10CFR55.43 b (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. The SRO is responsible approving discharges and determining the discharge process is complete.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		3
Group #		3
K/A #	2.3.12	
Importance Rating		3.8

2.3.12: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question: 99

The reactor is at power and primary containment entry is required to repair a fan coil unit. Containment de-inerting is in progress and the following data is noted:

Time	**Drywell Airborne Particulate**
1000	6400 CPM
1030	6050 CPM
1100	4950 CPM
1130	4700 CPM

When do containment conditions <u>first</u> allow entry into containment?

a. 1000

b. 1030

c. 1100

d. 1130

Answer:

c. 1100

Explanation (Optional):

Procedure 2.0.10, PRIMARY CONTAINMENT ACCESS CONTROL, Section 4, Step 4.7.1.1 requires drywell airborne particulate levels to be < 5000 cpm. The drywell coordinator must verify this with the Shift Manager.

Distracters:

a. is incorrect because particulate levels must be < 5000 cpm. The candidate that cannot recall limiting airborne particulate levels may select this answer. This answer is plausible because particulate levels are close to the required limits.

b. is incorrect because drywell particulate levels must be < 5000 cpm. The candidate that cannot recall limiting airborne particulate levels may select this answer. This answer is plausible because the particulate levels are close to the required limits.

d. is incorrect because the particulate level was adequate the prior 30 minutes. The candidate that cannot remember the particulate thresh hold may select this answer. This answer is plausible because the particulate limits are adequate for entry. The question is asking when the limits were first adequate for entry.

Technical Reference(s):
2.0.11, Rev 38 PRIMARY CONTAINMENT ACCESS CONTROL

Proposed references to be provided to applicants during examination: None

Learning Objective:
SKL0110101001370A 2.0.10, Primary Containment Access Control: Discuss the following as described in Conduct of Operations Procedure 2.0.10, Primary Containment Access Control: 2) Initial Primary Containment entry during power operations

Question Source:
Bank #
Modified Bank #
New X

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

10 CFR Part 55 Content: 55.41 _____
55.43 4

Comments:

Difficulty 3

SRO Only - 10CFR55.43 b (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. The SRO is required to be familiar with the requirements to enter the drywell when at power and must give permission for the entry to occur.

Examination Outline Cross-Reference:

Level	~~RO~~	SRO
Tier #		3
Group #		4
K/A #	2.4.47	
Importance Rating		4.2

2.4.47: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Question: 100

An accident is occurring resulting in a radiation release to the environment. **It is unknown at what time the release began.** The following are the known release rates and times following discovery of the release:

T=0 min	Dose assessment determines the release rate is above the Table A-1 level of an ALERT.
T=5 min	Dose assessment determines the release rate remains the same as at T=0.
T=15 min	Dose assessment determines the release rate is 10% higher than at T=0.
T=18 min	Dose assessment determines the release rate remains 10% higher than at T=0.

What is the LATEST time the Emergency Director is allowed to make the emergency declaration and adhere to the emergency plan?

a. T=0 min

b. T=15 min

c. T=18 min

d. T=30 min

Answer:

b. T=15 min

Explanation (Optional):

In accordance with Emergency Procedure 5.7.1 Emergency Classification; For Radiological Effluent Action Levels, NOTE 2 states: " 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. The bases for the release EAL is that the release has been above the EAL for > 15 minutes. 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. Because it is unknown at what time the release began the Emergency Director must assume the time limitation has elapsed when the first dose assessment reveals the limit for an ALERT is exceeded at time T=0

min. The Emergency Director has 15 minutes to declare once information is available revealing an EAL is met.

Distracters:

a. This is not correct because of Note 2. At T=0 the EAL is met and the ED has up to 15 minutes to declare. The candidate that does not realize the ED has up to 15 minutes to declare may select this answer. This answer is plausible because the EAL is met at this time.

c. This is not correct because of Note 1. At Time = 18 min, the release rate has increased and is still above the ALERT level. A candidate may choose this answer if they forget that Note 1 is there. This answer is plausible because there is a NOTE 1 that states to not wait for the declaration if it is determined the condition will likely exceed the applicable time.

d. This is not correct because of Note 1. At Time = 30 min, the release rate is still above the ALERT level. A candidate may choose this answer if they forget that Note 1 is there and figure the EAL has to be met for 15 minutes plus the ED is allowed 15 minutes to declare. This answer is plausible because the 15 minutes has elapsed plus the 15 minute allowable declaration time.

Technical Reference(s):
5.7.1, Rev 50, Emergency Classification

Proposed references to be provided to applicants during examination: EPIP Hard card Chapter A with accompanying NOTES.

Learning Objective:
5. Concerning event classification:
a. Given a copy of EPIP 5.7.1 and an EAL identification code, determine the EAL and its associated emergency classification.

Question Source:
Bank #
Modified Bank #
New X

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

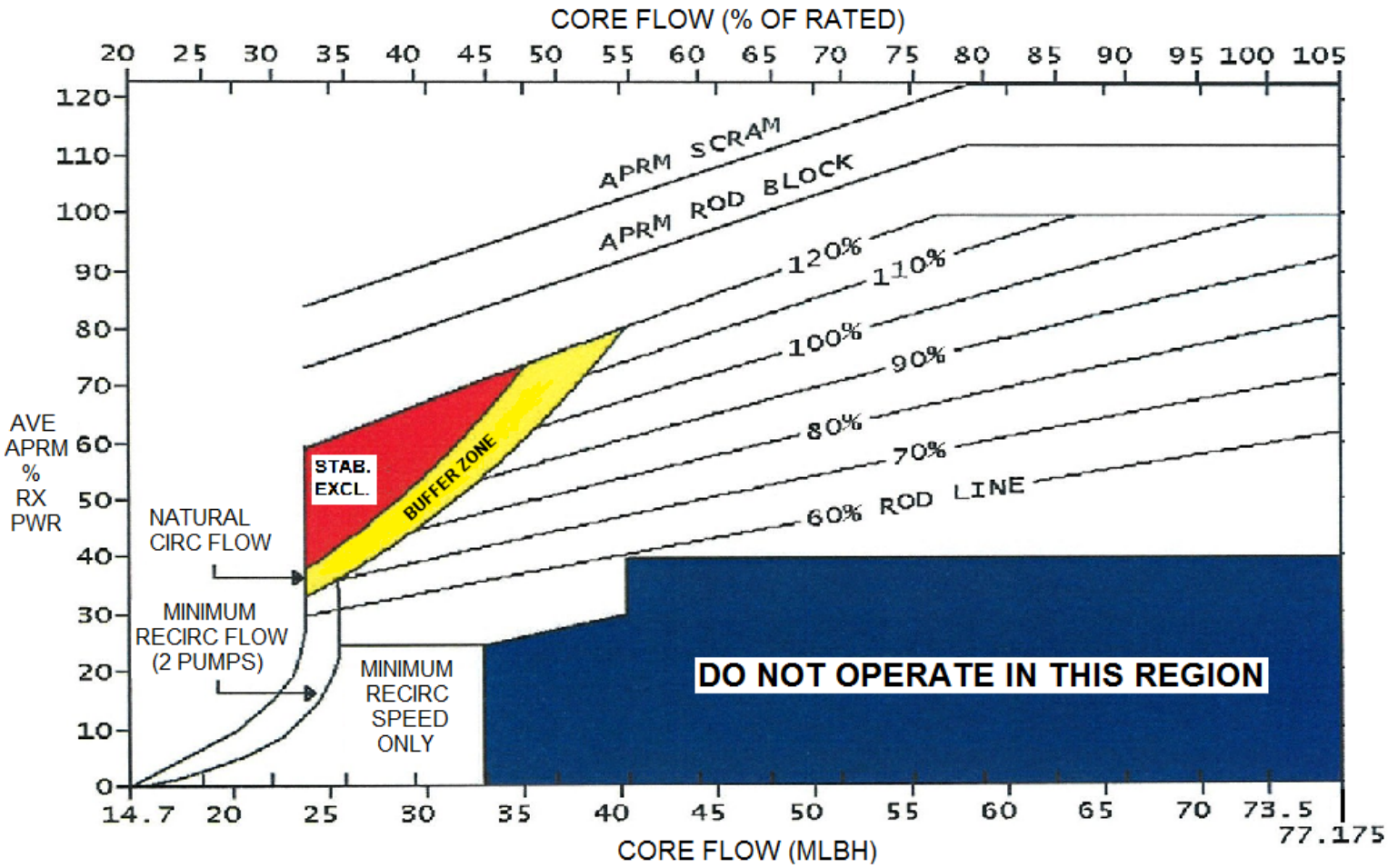
10 CFR Part 55 Content:
55.41
55.43 5

Difficulty 4

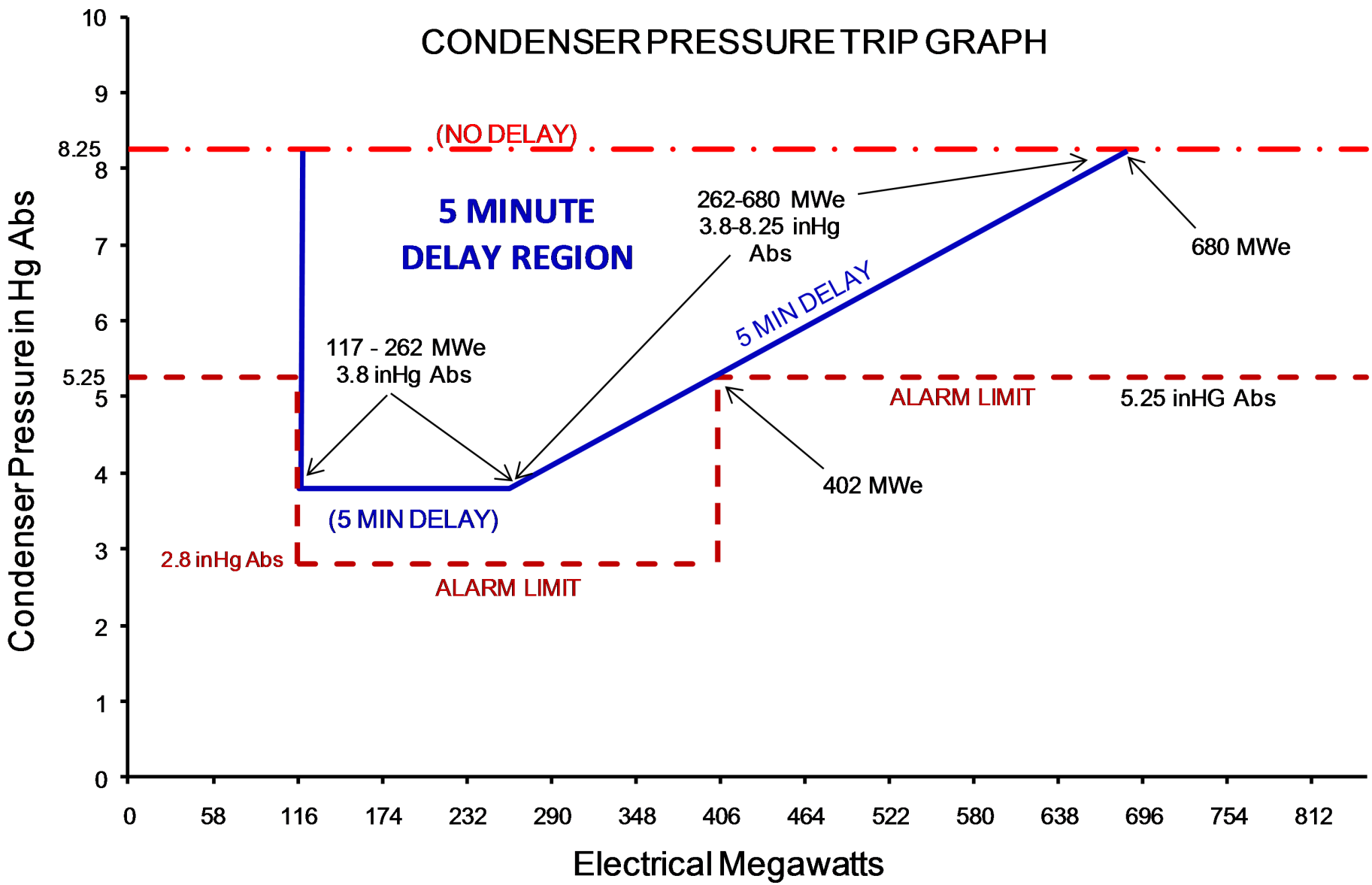
SRO Only - 10CFR55.43 b (5)Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The SRO is responsible for declaring EALs and determining when the EAL thresh hold is reached.

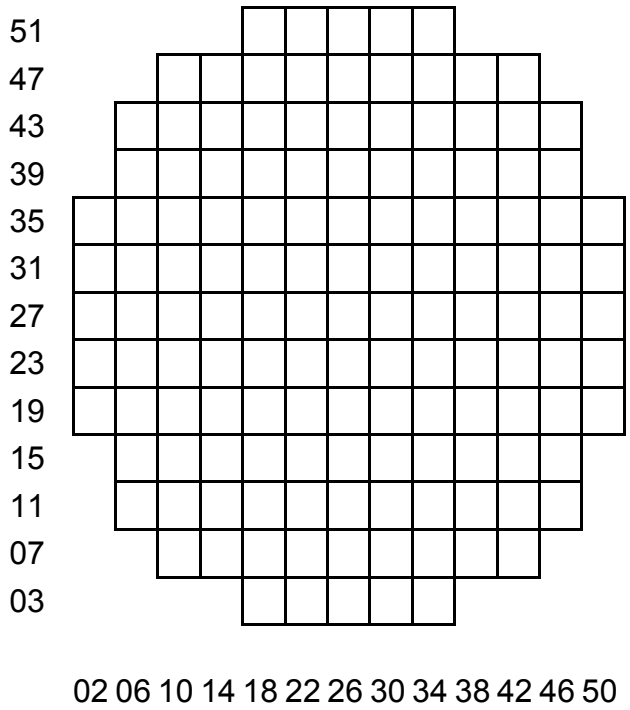
SRO Handouts

POWER TO FLOW MAP - CYCLE 28



NOTE – The Maximum Effective Load Line Limit (MELLL) is 118.9% RTP (line not shown on PMIS screen).


CONDENSER PRESSURE TRIP GRAPH
(NO DELAY)
5 MINUTE
DELAY REGION
262-680 MWe
3.8-8.25 inHg
Abs
680 MWe
5 MIN DELAY
117 - 262 MWe
3.8 inHg Abs
ALARM LIMIT
5.25 inHG Abs
402 MWe
(5 MIN DELAY)
2.8 inHg Abs
ALARM LIMIT
8.25
5.25
Condenser Pressure in Hg Abs
Electrical Megawatts
0
1
2
3
4
5
6
7
8
9
10
0
58
116
174
232
290
348
406
464
522
580
638
696
754
812


51
47
43
39
35
31
27
23
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03
02 06 10 14 18 22 26 30 34 38 42 46 50

Core Map

A — Abnorm. Rad Release / Rad Effluent

1 — Offsite Rad Conditions

AG1.1 1 | 2 | 3 | 4 | 5 | DEF

Any valid gaseous monitor reading > Table A-1 column "GE" for ≥ 15 min. (Note 1)

AG1.2 1 | 2 | 3 | 4 | 5 | DEF

Dose assessment using actual meteorology indicates doses > 1 Rem TEDE or > 5 Rem thyroid CDE at or beyond the site boundary

AG1.3 1 | 2 | 3 | 4 | 5 | DEF

Field survey results indicate closed window dose rates > 1 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1)
OR
Analyses of field survey samples indicate thyroid CDE > 5 Rem for 1 hr of inhalation at or beyond the site boundary

AS1.1 1 | 2 | 3 | 4 | 5 | DEF

Any valid gaseous monitor reading > Table A-1 column "SAE" for ≥ 15 min. (Note 1)

AS1.2 1 | 2 | 3 | 4 | 5 | DEF

Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem thyroid CDE at or beyond the site boundary

AS1.3 1 | 2 | 3 | 4 | 5 | DEF

Field survey indicates closed window dose rate > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1)
OR
Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary

AA1.1 1 | 2 | 3 | 4 | 5 | DEF

Any valid gaseous monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

AA1.2 1 | 2 | 3 | 4 | 5 | DEF

Any valid liquid effluent monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

AA1.3 1 | 2 | 3 | 4 | 5 | DEF

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM limits for ≥ 15 min. (Note 2)

AU1.1 1 | 2 | 3 | 4 | 5 | DEF

Any valid gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

AU1.2 1 | 2 | 3 | 4 | 5 | DEF

Any valid liquid effluent monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

AU1.3 1 | 2 | 3 | 4 | 5 | DEF

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODAM limits for ≥ 60 min. (Note 2)

2 — Onsite Rad Conditions & Spent Fuel Pool Events

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 µCi/sec	3.50E+07 µCi/sec	2.80E+06 µCi/sec	2.24E+05 µCi/sec
GASEOUS	Rx Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.45E+05 µCi/sec	8.48E+04 µCi/sec
GASEOUS	Turb Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.62E+05 µCi/sec	9.02E+04 µCi/sec
GASEOUS	RW / ARW Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.64E+05 µCi/sec	9.08E+04 µCi/sec
LIQUID	Rad Waste Effluent	-----	-----	The lesser of *: 200 x calculated alarm values OR monitor upscale	The lesser of *: 2 x calculated alarm values OR monitor upscale
LIQUID	Service Water Effluent	-----	-----	4.80E-04 µCi/cc	4.80E-06 µCi/cc

* with effluent discharge **not** isolated

AA2.1 1 | 2 | 3 | 4 | 5 | DEF

Damage to irradiated fuel **OR** loss of water level (uncovering irradiated fuel outside the RPV) that causes **EITHER** of the following:
Valid RMA-RA-1 Fuel Pool Area Rad reading > 50 R/hr
OR
Valid RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm

AA2.2 1 | 2 | 3 | 4 | 5 | DEF

A water level drop in the reactor refueling cavity or spent fuel pool that will result in irradiated fuel becoming uncovered

AU2.1 1 | 2 | 3 | 4 | 5 | DEF

Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by **any** of the following:
- LI-86 (calibrated to 1001' elev.)
- Spent fuel pool low level alarm
- Visual observation

AND
Valid area radiation monitor reading rise on RMA-RA-1 or RMA-RA-2

AU2.2 1 | 2 | 3 | 4 | 5 | DEF

Unplanned valid area radiation monitor reading or survey results rise by a factor of 1,000 over normal levels*

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

3 — MCR/CAS Rad

AA3.1 1 | 2 | 3 | 4 | 5 | DEF

Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:
Main Control Room (RM-RA-20)
OR
CAS

Notes

1. 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, declaration should be based on dose assessment instead of radiation monitor values.
   (See EAL AS1.2/AG1.2.) Do **not** delay declaration awaiting dose assessment results.
2. 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.
3. 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.
4. Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.
5. Manual scram methods for EAL SA2.1 and EAL SS2.1 are the following:
   - Reactor Scram push buttons
   - Reactor Mode switch in SHUTDOWN
   - Manual or auto actuation of ARI
6. See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage.
7. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should **not** be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.
8. The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an offsite power source.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	----------NOTE---------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. ----------	
	A.1 Verify stuck control rod separation criteria are met	Immediately
	AND	
	A.2 Disarm the associated control rod drive (CRD).	2 hours
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM.
	AND		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	----------NOTE---------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ----------	
		Fully insert inoperable control rod.	3 hours
	AND		
	C.2	Disarm the associated CRD.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ----------NOTE---------- Not applicable when THERMAL POWER > 9.85% RTP. ---------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with BPWS. OR D.2 Restore control rod to OPERABLE status.	4 hours 4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 10 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------

During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	200 days cumulative operation in MODE 1
SR 3.1.4.3	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the affected core cell AND Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

----------NOTES----------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[a][b] (seconds) when REACTOR STEAM DOME PRESSURE ≥ 800 psig
46	0.44
36	1.08
26	1.83
06	3.35

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig, are within established limits.

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
	OR		
	A.2	Place associated trip system in trip.	12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	OR		
	B.2	Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < 29.5% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

3.3 INSTRUMENTATION

3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3.1 The PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each Function. For Function 5, separate Condition entry is allowed for each penetration flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.6.	Immediately
C. One or more Functions with two required channels inoperable. OR One Function 2.c channel inoperable.	C.1 Restore one required channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.3.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	E.1 Be in MODE 3.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	F.1 Initiate action in accordance with Specification 5.6.6.	Immediately

T 3.3 INSTRUMENTATION

T 3.3.1 Control Rod Block Instrumentation

TLCO 3.3.1 The control rod block instrumentation for each Function in Table T3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table T3.3.1-1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with less than minimum channels OPERABLE.	A.1 Initiate Reactor Manual Control System rod withdrawal block.	1 hour

D 3.1 LIQUID EFFLUENTS

D 3.1.2 Liquid Waste Concentration

DLCO 3.1.2 The concentration of radioactive materials in liquid wastes from pre-release analysis shall be ≤ 0.01 µCi/ml, excluding tritium and noble gases.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of radioactive materials in liquid wastes from pre-release analysis > 0.01 µCi/ml, excluding tritium and noble gases.	A.1 Appropriate parts of the liquid radwaste treatment system shall be used to reduce the concentration.	Prior to liquid waste discharge
B. Required Action and associated Completion Time not met. AND Radioactive liquid waste being discharged without treatment in excess of 0.01 µCi/ml, excluding tritium and noble gases.	B.1 Prepare and submit a Special Report to the NRC pursuant to Specification D 5.4 that identifies equipment or subsystems not OPERABLE and the reason for the inoperability, action(s) taken to restore the inoperable equipment to OPERABLE status and a summary description of the action(s) taken to prevent a recurrence.	31 days following the end of the quarter in which the limit was exceeded

D 3.1 LIQUID EFFLUENTS

D 3.1.3 Liquid Effluents Dose

DLCO 3.1.3 The dose to a Member of the Public due to radioactive material in liquid effluents beyond the SITE AND EXCLUSION AREA BOUNDARY (Figure D2.a-1) shall be limited to:

a. ≤ 1.5 mrem to the total body or ≤ 5.0 mrem to any body organ during any calender quarter; and

b. ≤ 3.0 mrem to the total body or ≤ 10.0 mrem to any body organ during any calender year.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated dose due to radioactive material in liquid effluents beyond the SITE AND EXCLUSION AREA BOUNDARY exceeds the limit.	A.1 Prepare and submit a Special Report, in lieu of any other report, pursuant to Specification D 5.4 to the NRC which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken.	31 days following the end of the quarter in which the limit was exceeded

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Calculated dose due to radioactive material in liquid effluents beyond the SITE AND EXCLUSION AREA BOUNDARY exceeds two times the limit.	B.1 Prepare and submit a Special Report, in lieu of any other report, pursuant to Specification D 5.4 to the NRC which: 1) defines actions to be taken to reduce releases and prevent recurrence, and 2) results of an exposure analysis including effluent pathways and direct radiation to determine whether the dose or dose commitment to a Member of the Public due to radiation and radioactive releases from Cooper Station during the calender year through the period covered by the calculation was ≤ 75 mrem to the thyroid and ≤ 25 mrem to the total body and all other body organs.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.1.3.1	Perform an assessment of compliance with DLCO 3.1.3.	31 days
DSR 3.1.3.2	Project a prospect of compliance with DLCO 3.1.3 for radioactive liquid releases without radwaste system in operation.	In any quarter in which Radioactive liquid releases are made and the radwaste system is not operated

ATTACHMENT 1 LIQUID RADIOACTIVE WASTE DISCHARGE FORM

Section 1. REQUEST FOR ANALYSIS OF RADIOACTIVE LIQUID WASTE PRIOR TO DISCHARGE

To: Chemistry From: Shift Manager Tank To Be Discharged: _____

Started Recirculation For Sample: Time: _____ Date: _____

Recirculation Of Tank Complete: Time: _____ Date: _____

Estimated Volume To Be Discharged: _____

Shift Manager: _____ Time: _____ Date: _____

Section 2. THIS SECTION TO BE COMPLETED BY PERSON TAKING SAMPLE

Monitor Source Check

Informed Control Room And Performed Source Check Initials: _____

Monitor Background: _____ Monitor Source Check Value: _____

Sample Point: _____ Time: _____ Date: _____

Signature: _____

Section 3. AUTHORIZATION TO RELEASE RADIOACTIVE LIQUID WASTE

To: Shift Manager From: Chemistry Release Authorization Number: _____

Total μCi/ml: _____

Total Concentration is < 1.0E-02 μCi/ml YES/NO

Signature: _____

31 Day Dose, Percent Of Annual Limit For Each Value Is ≤ 2.0E+00 YES/NO

Signature: _____

You Are Authorized To Release Subject Tank With Either Of Following Restrictions:

Maximum Liquid Waste Discharge Rate (gpm)

1) _____ 2) _____ 3) _____

Minimum Dilution Flow To Canal (gpm)

1) 159,000 2) 159,000 3) 159,000

Discharge Monitor Alarm Setpoint (μCi/ml)

1) _____ 2) _____ 3) _____

NOTE – Terminate Discharge If Above Specifications Cannot Be Maintained. Contents Of This Tank Are Within Chemical Parameters For Discharge.

Chemistry: _____ Time: _____ Date: _____

Section 4. SHIFT MANAGER APPROVAL TO RELEASE

4.1 Circle Appropriate Discharge Canal Flow Rate:

NUMBER OF OPERATING CW PUMPS	AVERAGE CW DISCHARGE FLOWRATE (gpm)	
	DE-ICING	NO DE-ICING
4	378,600	631,000
3	308,400	514,000
2	193,200	322,000
1	118,800	198,000

4.2 To: Operations Personnel From: Shift Manager

The Subject Tank Contents Are Approved For Release Subject To The Following Restrictions:

1) Maximum Liquid Disch Rate: _____ gpm (Section 3)

2) Minimum Dilution Flow To Canal Of: _____ gpm (Section 3)

3) Alarm Limits Specified (Section 3)

   2 x Alarm Limit: _____

   200 x Alarm Limit: _____

4) Tank Volume Verified: _____ (Compare To Section 1)

5) DISCHARGE IN PROGRESS Tags Installed On Running Circ Water Pumps.

Approval To Release:

Shift Manager: _____ Time: _____ Date: _____

Section 5. To: Chemist From: Shift Manager

The Subject Discharge Has Been Completed And The Following Data Obtained During The Discharge:

START RUN NUMBER	DATE	TIME	% TANK LEVEL	STOP RUN NUMBER	DATE	TIME	% TANK LEVEL
1				1			
2				2			
3				3			
4				4			

Usable Tank Volumes (0% to 100%)

FDST - 18,707 gal

WST A - 20,015 gal

WST B - 20,015 gal

1) Dilution Flow (Section 4.1): _____ gpm

2) Volume Of Release: _____ gal

3) Total Time Of Discharge: _____ min

4) Discharge Flow: _____ gpm

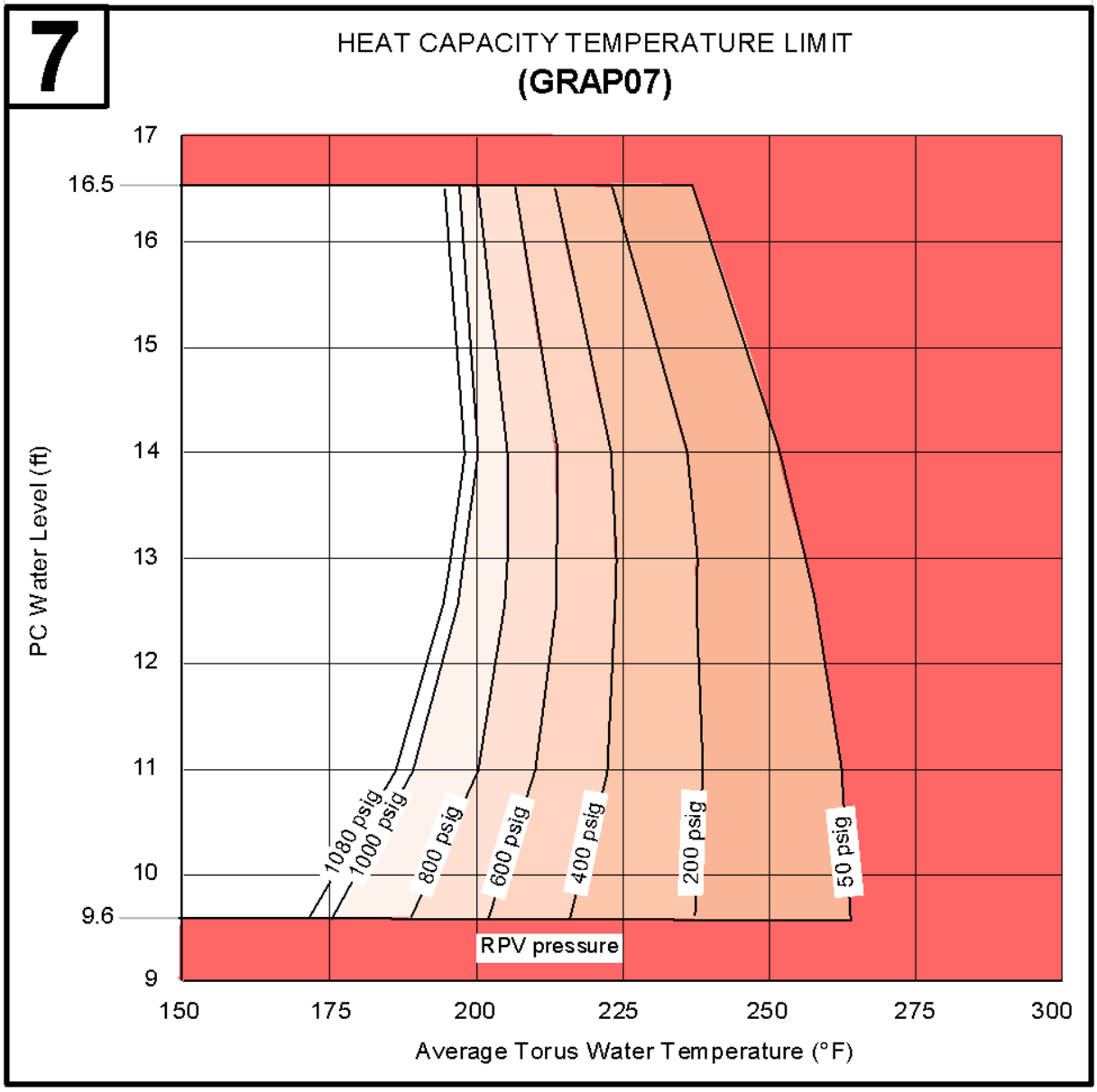
5) River Level: _____ ft MSL

6) Remove DISCHARGE IN PROGRESS Tags From Running Circ Water Pumps.

7) Total River Flow: _____ l/min
(Determined By The Chemist)

Shift Manager: _____ Time: _____ Date: _____

Chemist Review: _____ Date: _____


7
HEAT CAPACITY TEMPERATURE LIMIT
(GRAP07)
PC Water Level (ft)
17
16.5
16
15
14
13
12
11
10
9.6
9
150
175
200
225
250
275
300
Average Torus Water Temperature (°F)
1080 psig
1000 psig
800 psig
600 psig
400 psig
200 psig
50 psig
RPV pressure