Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Change in leak rate with change in pressure	**Tier #**	1
	Group #	1
	K/A #	AK1.02
	Rating	3.1

Question 1

GIVEN:

- A Pressurizer PORV fails open and cannot be isolated
- The plant trips and SI actuates
- 30 minutes after the reactor trip the crew enters E-1.2, Post LOCA Cooldown and Depressurization

Which of the following describes the expected plant conditions as the crew enters E-1.2?

A. Break flow is unchanged from its original value; Pressurizer level off-scale high.

B. Break flow has decreased from its original value; Pressurizer level on scale and decreasing.

C. Break flow is unchanged from its original value; Pressurizer level on scale and decreasing.

D. Break flow has decreased from its original value; Pressurizer level off-scale high.

Proposed Answer: D.

Explanation:

A. Incorrect – RCS pressure will be less than NOP as a result break flow will be reduced.

B. Incorrect – Pressurizer level will be off scale high (even if still on scale, it would be increasing due to increased SI flow and the open PORV).

C. Incorrect – Break flow will be reduced.

D. Correct – As RCS pressure lowers, the break flow rate will lower and due to the failed open PORV, pressurizer level will be off scale high.

Technical Reference(s): LMCD-FRC page 17 and page 37

Proposed references to be provided to applicants during examination: None

Learning Objective: 41697 - Describe the plant response to a loss of reactor coolant including: Vapor Space LOCAs

Question Source: Bank # DC 2010-01 Question #39
Modified Bank #

Question History: Last NRC DC 2010-01

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

10 CFR Part 55 Content: 55.41 5 / 55.43

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between the small break LOCA and the Steam Generators	**Tier #**	1
	Group #	1
	K/A #	EPE 009 EK2.03
	Rating	3.0

Question 2

GIVEN:

- A Small Break LOCA has occurred in Containment
- RCS pressure is 1500 psig and slowly lowering
- Only Train 'A' of ECCS has actuated
- All MSIVs are closed
- All RCPs are running
- All steam generators are available

A secondary heat sink ______ required to maintain adequate core cooling because _____.

A. is not; core cooling is provided by the operation of the RCPs

B. is not; core cooling is provided the cooling effect of the lowering RCS pressure.

C. is; core cooling is provided by the secondary heat sink to allow ECCS flow to equal break flow

D. is; secondary heat sinks are required for all LOCA events

Proposed Answer: C

Explanation:

For breaks in this category, the establishment of an equilibrium pressure where pumped SI equals break flow constitutes a safe and stable condition for the long term, provided that the steam generator heat sink is maintained until such time that the break flow and SI sensible heat can remove all the decay heat._ Once equilibrium pressure was established, the core was covered and adequate flow existed to remove decay heat through the steam generator with a small amount of voiding. The only change in the primary system conditions through the transient for these cases is a gradual decrease in fluid temperatures which is beneficial, since it indicates that adequate core cooling is being maintained.

A. Incorrect – RCPs only transport heat from the core to the steam generators

B. Incorrect – a secondary heat sink is required to remove sufficient heat to lower primary pressure to allow for ECCS flow to equal leak rate.

C. Correct core cooling is provided by the secondary heat sink to allow ECCS flow to equal break flow

D. Incorrect – secondary heat sink is not required for a LBLOCA due to the primary completely depressurizes during the event

Technical References: WOG Background Information, E-1 Loss of Reactor or Secondary Coolant, Revision 2
LPE-1A – Loss of Coolant Response, Revision 13

References to be provided to applicants during exam: None

Learning Objective: 7920 - Explain basis of emergency procedure step

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for the following responses as the apply to the Large Break LOCA: **Hot-leg injection/recirculation**	**Tier #**	1
	Group #	1
	K/A #	EK3.13
	Rating	3.8

Question 3

GIVEN:

- A LOCA occurred on Unit 1 about 7 hours ago
- RCS pressure is 25 psig
- Procedure EOP E-1.3, Transfer to Cold Leg Recirculation, is in effect.

Based on these conditions, the next operator action and the reason for it is to:

A. Transfer to Hot Leg Recirculation to mitigate the effects of boric acid precipitation in the core.

B. Maintain Cold Leg Recirculation until the Refueling Water Storage Tank level is less than 4% to maximize the water inventory in the Containment sump.

C. Transfer to Hot Leg Recirculation to mitigate the effects of boric acid precipitation in the core.

D. Transfer to Hot Leg Recirculation to prevent a reverse flow through the core as the flow rate out of the break exceeds the injection flow rate.

Proposed Answer: C.

Explanation:

A. Incorrect. During Cold leg recirculation, core boil-off will concentrate boric acid in the core region. Hot leg recirculation will remix boron to a more even distribution in the cooling water flow. This mitigates the effects of boric acid precipitation in the core.

B. Incorrect. Swapover to Hot Leg Recirc begins at 6.5 hours after the LOCA, per EOP E-1, step 20, Loss of Reactor or Secondary Coolant, transition to Hot Leg Recirculation (and is aligned after 7 hours).

C. Correct. Hot Leg Recirculation prevents boric acid precipitation and quenches the steam bubble in the Reactor vessel head.

D. Incorrect. Hot Leg Recirculation establishes a reverse flushing flow through the core.

Technical References: LPE-1C pages 2,4,5,18; LMCDFRC pages 4,16; EOP E-1 pages 1,19; EOP 1.4 page 1.

References to be provided to applicants during exam: None

Learning Objective: 8914-Explain the reason for transferring to hot leg recirculation after a LOCA.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR 41.5 / 41.10 / 45.6 / 45.13	

Examination Outline Cross-Reference	Level	RO
Ability to determine and interpret when to secure RCPs on high stator temperature as it applies to Reactor Coolant Pump Malfunctions (Loss of RC Flow).	**Tier #**	1
	Group #	1
	K/A #	APE 015/17 AA2.09
	Rating	3.4

Question 4

The plant is operating at 100% power. Annunciator Response Procedure AR PK05-01, RCP NO. 11, directs the operator to manually trip the reactor and RCP 11, if not already tripped, after getting Shift Foreman concurrence if:

A. Motor bearing temperature is 175°F.

B. Seal water differential pressure is 275 psid.

C. Motor stator temperature is 310°F.

D. Seal outlet temperature is 200°F.

Proposed Answer: **C**

Explanation:

A. Incorrect – A trip is required if motor bearing temperature is greater than 200 degrees.

B. Incorrect - A trip is required if seal water differential pressure is less than 255 psid.

C. Correct – In accordance with AR PK05-01, Section 2.7, "RCP 1-1 High Temperature PPC," requires that if the temperature is at or above 300°F then trip the RCP following manual trip of the reactor.

D. Incorrect - A trip is required if seal outlet temperature is greater than 235°F.

Technical References: AR PK05-01, RCP No. 11, Revision 33
LAR-1, RCP Failures, Revision 11

References to be provided to applicants during exam: None

Learning Objective: 7927 - Given initial conditions and assumptions, determine if a reactor trip or safety injection is required.

Question Source: Bank #
(note changes; attach parent) Modified Bank #
New X

Question History: Last NRC Exam NA

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. 000022 Loss of Rx Coolant Makeup / 2	**Tier #**	1
	Group #	1
	K/A #	022 2.4.21
	Rating	4.0

Question 5

GIVEN:

- Unit 1 has experienced a SBLOCA from one of the Reactor Coolant Pump Seals
- All rods are fully inserted
- Reactor Power is approximately 0.7% with a negative IR SUR
- Core Exit Thermocouples indicate 950 degrees
- Core Exit Subcooling Margin is 27ºF
- Containment Radiation Monitors 4.3 R/HR
- Pressurizer Level is 14%

Based on these conditions, which safety function, if any, is in jeopardy?

A. Core Cooling

B. Subcriticality

C None, all Safety Functions are met

D. Inventory

Proposed Answer: D.

Explanation:

A. Incorrect. F-0.2 criteria, Core Exit Thermocouples are less than 1200ºF and SCM is greater than 20ºF.

B. Incorrect. F-0.1 criteria, Power Range power is less than 5% and IR SUR is ≤ 0.

C. Incorrect. F-0.6 criteria are not fully met.

D. Correct. F-0.6 criteria, Pressurizer level is <17%. EOP FR-I.2 should be entered.

Technical References: LPA-19, EOP F-0 attachments 6.1-6.6, EOP FR-I.2

References to be provided to applicants during exam: None

Learning Objective: 3478 - State the entry conditions for abnormal operating procedures.

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X

Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.7 / 43.5 / 45.12	

Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor the LPI pump control switch, indicators, ammeter, running light and flow meter as they apply to the Loss of Residual Heat Removal System.	**Tier #**	1
	Group #	1
	K/A #	APE 025 AA1.09
	Rating	3.2

Question 6

GIVEN:

- Unit 1 entered Mode 6 five days ago
- RHR A is in shutdown cooling
- RHR A Cold Leg flow indicates 700 GPM flow

Which of the following would be indicative of According to OP AP SD-5, Loss of Residual Heat Removal, a loss of Residual Heat Removal would be indicated by an RHR pump motor current of:

A. 30 amps with FCV-641A, RHR A Pump Recirc Valve, Open.

B. 30 amps with FCV-641A, RHR Pump A Recirc Valve, Closed.

C. 25 amps with FCV-641A, RHR Pump A Recirc Valve, Open.

D. 25 amps with FCV-641A, RHR Pump A Recirc Valve, Closed.

Proposed Answer: C

Explanation:

A. Incorrect –An RHR pump motor current >28 Amps with FCV-641A/B fully Open is consistent with RHR flow in accordance with OP AP SD-5 and FCV-641A fully opens at 728 GPM flowrate.

B. Incorrect – RHR pump motor current >28 Amps with FCV-641A/B fully Open is consistent with RHR flow in accordance with OP AP SD-5 but FCV-641A/B would be fully open given these conditions (700 GPM flowrate). FCV-641A/B will close if flow increases above 1398 GPM or associated RHR pump is secured.

C. Correct – RHR pump motor current ≤28 Amps with FCV-641A/B Fully Open is indications of loss of residual heat removal flow in accordance with OP AP SD-5.

D. Incorrect - RHR pump motor current ≤28 Amps with FCV-641A/B Fully Open is indications of loss of residual heat removal in accordance with OP AP SD-5. In addition, FCV-641A/B would be fully open given these conditions (700 GPM flowrate). FCV-641A/B will close if flow increases above 1398 GPM or associated RHR pump is secured.

Technical References: LB-2, Residual Heat Removal System, Revision 16 OP AP SD-5, Loss of Residual Heat Removal, Revision 9A

References to be provided to applicants during exam:		None
Learning Objective:		
Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/**Fundamental** **Comprehensive**/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: RCP injection flow	Tier #	1
	Group #	1
	K/A #	AA2.14
	Rating	2.8

Question 7

GIVEN:

- Unit 2 is at 100% power
- The Pressurizer Channel Selector Switch is in the PT-455/PT-456 position
- Charging is in MANUAL

PT-455 fails high

Based on these conditions, initially RCP seal injection flow will:

A. decrease due to a lower differential pressure across HCV-142.

B. increase due to a higher differential pressure across HCV-142.

C. increase due to a lower differential pressure across HCV-142.

D. decrease due to a higher differential pressure across HCV-142.

Proposed Answer: B.

Explanation:

A. Incorrect. Seal Injection is supplied by the CVCS System at 8 to 13 gpm per RCP. The flow rate is normally adjusted by throttling HCV-142 to divert charging flow to the seals. The purpose of HCV-142 is to create sufficient backpressure in the charging line to ensure that adequate flow is maintained through the RCP seal water injection line upstream of valve HCV-142. PT-455 failing high opens the spray valves (and two PORVs), which come off of cold legs 1 and 2. When spray is initiated charging injection to the cold legs increases and pressure on the charging pump side of HCV-142 lowers; thereby changing the backpressure of HCV-142. This lowers raises RCP seal injection until HCV-128 is adjusted.

B. RCP seal injection will increase, because Differential pressure across HCV-142 will decrease.

C. Incorrect – See A.

D. Incorrect – Differential pressure across HCV-142 will decrease not increase.

Technical References: LB-1A page 43, LA-6 page 13, OIM A-4-4B Rev 24, OIM B-1-1 Rev 27

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/**Analysis**	 X
10CFR Part 55 Content:	CFR: 43.5 / 45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operation implications of the following concepts as they apply to ATWS: Effects of boron on reactivity	**Tier #**	1
	Group #	1
	K/A #	EPE 029 EK1.03
	Rating	3.6

Question 8

While taking actions in accordance with FR-S.1, Response to Nuclear Power Generation, ATWS, the reason boron is added to the reactor is to counter the _________________ following an ATWS.

A. decrease in core temperature

B. decrease in reactor coolant temperature

C. increase in RCS pressure

D. rods not being fully inserted

Proposed Answer: D

Explanation:

A. Incorrect – Decrease in core temperatures result in a positive reactivity addition but the concern is the rods and the SDM

B. Incorrect - Decrease in the reactor coolant temperatures will result in a positive reactivity addition but the concern is the rods and the SDM.

C. Incorrect – Increase in RCS pressure result in a negative reactivity addition verses a positive reactivity addition

D. Correct – Boron addition provides negative reactivity to ensure shutdown margin is established following a ATWS

Technical References: EOP E-0, Reactor Trip or Safety Injection, Rev. 31
EOP FR-S.1, Response to Nuclear Power Generation / ATWS, Rev. 13A
LCMDFRS, Mitigating Core Damage – Subcriticality, Revision 6
WOGBD, EOP FR-S.1, Response to Nuclear Power Generation / ATWS, Revision 2

References to be provided to applicants during exam: None

Learning Objective: 8904 – State the means to verify the reactor shutdown

Question Source: Bank #
(note changes; attach parent) Modified Bank #
New X

Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.8	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between the Steam Line Rupture and the following: Valves	Tier #	1
	Group #	1
	K/A #	AK2.01
	Rating	2.6

Question 9

A safety injection and reactor trip has occurred due to low steam pressure.

GIVEN;

- A Main Steam Isolation has actuated
- Steam Generator Pressures are:
  - 11 – 925 psig, stable
  - 12 – 310 psig, decreasing
  - 13 – 330 psig, decreasing
  - 14 – 950 psig, stable
- RCS Tave is decreasing

The actions of EOP E-0 have been completed through step 10. EOP E-2, Faulted Steam Generator Isolation, is the procedure in effect.

Which of the following is a possible cause of the given plant conditions?

A. Steam Break upstream of FCV-37, SG 1-2 Steam Supply to TDAFW Pump.

B. Steam Break upstream of TDAFW Steam Supply Valve FCV-95.

C. Main Steam Line 1-2 safety valve, RV-7, has lifted and is stuck open.

D. Steam Break downstream of Steam Generator 1-3 MSIV FCV-43.

Proposed Answer: B

Explanation:

A. Incorrect. Both SGs 1-2 and 1-3 are depressurizing. Check valves between the generators (off of the AFW pump line) prevent both from depressurizing.

B. Correct. Both SGs 1-2 and 1-3 feed the TDAFW pump. A break upstream of FCV-95 will depressurize both.

C. Incorrect. Both SGs 1-2 and 1-3 are depressurizing. Check valves between the generators (off of the AFW pump line) prevent both from depressurizing.

D. Incorrect. MSIVs are shut, which would have isolated the steam break if this were the location.

Technical References: EOP E-2, OIM C-2-1, OIM B-6-10, OIM B-6-5, OIM B-6-6, LC-2A pages 19-20, 22-23

References to be provided to applicants during exam:		None
Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR 41.7 / 45.7	

***Licensee recommended I try to merge my question with #48 from the April 2007 exam.

48.
The plant trips from full power due a steam break.
Plant conditions 5 minutes later:
• SI actuated
• MSI actuated
• FWI actuated
• All equipment operated as designed
• Steam Generator Pressures:
o 11 – 900 psig, stable
o 12 – 300 psig, decreasing
o 13 – 300 psig, decreasing
o 14 – 870 psig, stable.
Which of the following is a possible location for the steam break?

A. Upstream of Steam Generator 12 MSIV FCV-42.

B. Downstream of Steam Generator 13 MSIV FCV-43.

C. On the line to the TDAFW pump, upstream of FCV-95.

D. On the line to the TDAFW pump, downstream of FCV-95.

Examination Outline Cross-Reference	Level	RO
Ability to operate and monitor the following as they apply to a Station Blackout: Battery approaching fully discharged	**Tier #**	1
	Group #	1
	K/A #	EPE 055 EA1.05
	Rating	3.3

Question 10

Following a station blackout, the vital 125VDC batteries have been supplying power for 1.5 hours.

As the batteries become exhausted, the battery's discharge rate (in amps) will:

A. go down steadily until the design battery capacity is exhausted.

B. be fairly constant until the design battery capacity is exhausted, then go down rapidly.

C. go up steadily until the design battery capacity is exhausted.

D. be fairly constant until the design battery capacity is exhausted, then rapidly go up rapidly.

Proposed Answer: C

Explanation:

A. Incorrect

B. Incorrect

C. Correct – As the battery discharges, voltage will go down; hence current will go up given a constant power load. The trend will continue until the battery is completely discharged.

D. Incorrect

Technical References:

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X

10CFR Part 55 Content: 55.41.7

Examination Outline Cross-Reference	Level	RO
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation: **000056 Loss of Off-site Power / 6**	**Tier #**	1
	Group #	1
	K/A #	2.1.7
	Rating	4.4

Question 11

Unit 1 is at 100% power when a loss of off-site power occurs. Only emergency diesel generator 1-3 starts and supplies its vital 4 KV bus.

Which of the following equipment would have power?

A. Condensate Booster Pump 1-1 and Component Cooling Water Pump 1-3

B. Component Cooling Water Pump 1-3 and Auxiliary Feedwater Pump 1-3

C. Reactor Heat Removal Pump 1-1 and Centrifugal Charging Pump 1-2

D. Component Cooling Water Pump 1-1 and Auxiliary Saltwater Pump 1-1

Proposed Answer: D.

Explanation:

A. Incorrect. Condensate Booster Pump 1-1 is powered from nonvital bus D. CCW Pump 1-3 is powered from Bus H, which is powered from DG 1-1.

B. Incorrect. CCW Pump 1-3 is powered from Bus H, which is powered from DG 1-1.

C. Incorrect. RHR Pump 1-1 and CCP 1-2 are powered from Bus G, which is energized from DG 1-2.

D. Correct. CCW Pump 1-1 and ASW Pump 1-1 are powered from Bus F, which is energized by DG 1-3.

Technical References: LJ-6A 4KV System, AOP-26 Loss of Offsite Power, LPA-26 Loss of Offsite Power, OIM J-1-1

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the 4 kV System. (41081)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content: CFR: 41.5 / 43.5 / 45.12 / 45.13

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus	**Tier #**	1
	Group #	1
	K/A #	APE 057 AK3.01
	Rating	4.1

Question 12

Unit 2 is at 30% power.

Power is lost to PY-21, Vital AC Instrument Bus. Among the instruments that lost power are Power Range channel, N-41, and Turbine Impulse Pressure channel, PT-505.

Which of the following describes why the operators must place Rod Control in MANUAL?

A. PT-505 failure is causing rods to insert.

B. NI-41 failure is causing rods to insert.

C. PT-505 failure is causing rods to withdraw.

D. NI-41 failure is causing rods to withdraw.

Proposed Answer: A

Explanation:

A. PT-505 low causes a Tave/Tref mismatch. Rods begin to insert in an attempt to match Tave with the failed low Tref.

B. NI-41 fails low, but does not cause rods to move (auctioneered high).

C. PT-505 fails low, the mismatch is the opposite.

D. NI-41 failing low, could think that rods will withdraw to match turbine with reactor power. In fact, the failing low of the NI will not affect rod control.

References to be provided to applicants during exam: None

Learning Objective: 4274 – Explain the consequences of loss of vital instrument bus

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: **Actions contained in EOP for loss of dc power**	**Tier #**	1
	Group #	1
	K/A #	AK3.02
	Rating	4.0

Question 13

The reason the operator must ensure the main feedwater pumps are immediately tripped following a loss of DC bus 12 is:

A. in anticipation of AFW being in service following a reactor trip.

B. To prevent the feed pumps from damage due to over speeding.

C. to prevent over-pressurization of the feedwater system.

D. To prevent over filling the steam generators.

Proposed Answer: C.

Explanation:

A. Incorrect. A Caution in OP AP-23 states, “MFW pump recirc valves fail closed and MFP trip solenoids lose power, immediate action is required to runback and locally trip running MFPs to prevent over-pressurization.”

B. Incorrect. A Caution in OP AP-23 states, “MFW pump recirc valves fail closed and MFP trip solenoids lose power, immediate action is required to runback and locally trip running MFPs to prevent over-pressurization.”

C. Correct. A Caution Note in OP AP-23 gives the following information, “MFW pump recirc valves fail closed and MFP trip solenoids lose power, immediate action is required to runback and locally trip running MFPs to prevent over-pressurization.”

D. Incorrect. A Caution in OP AP-23 states, “MFW pump recirc valves fail closed and MFP trip solenoids lose power, immediate action is required to runback and locally trip running MFPs to prevent over-pressurization.”

Technical References: OP AP-23 Loss of Vital DC Bus, EOP E-0 Reactor Trip or Safety Injection, Lesson LPA-23 Loss of Vital DC Bus, ARP PK20-18 125 VDC Bus 11, 12, or 13 Undervoltage

References to be provided to applicants during exam: None

Learning Objective: Given an abnormal condition, summarize the major actions of the abnormal operating procedure to mitigate an event in progress. (3477)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X

Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR 41.5,41.10 / 45.6 / 45.1	

Examination Outline Cross-Reference	Level	RO
Ability to identify and interpret diverse indications to validate the response of another indication: Loss of Nuclear Service Water (ASW)	**Tier #**	1
	Group #	1
	K/A #	APE 062 2.1.45
	Rating	4.3

Question 14

The running ASW Pump trips and the Standby ASW Pump fails to start.

Which of the following annunciators would the operator would expect to alarm?

A. PK01-01, ASW Sys HX Delta P/HDR Press, and PK01-03, Aux Salt Water Pumps, are illuminated

B. PK01-01, ASW Sys HX Delta P/HDR Press, and PK-01-02, Aux Salt Water PPS Room, are illuminated

C. PK-01-02, Aux Salt Water PPS Room, and PK13-01, Bar Racks/Screens, are illuminated

D. PK01-03, Aux Salt Water Pumps, and PK13-01, Bar Racks/Screens, are illuminated

Proposed Answer: A

Explanation:

A. Correct – PK01-01 actuates on ASW to CCW HX low pressure and PK01-03 actuates on a pump failure

B. Incorrect - PK01-02 does not actuate on a loss of ASW Pump. Actuates on Aux Salt Water Pump Room adverse conditions.

C. Incorrect - PK01-02 does not actuate on a loss of ASW Pump and PK13-01 does not actuate on low ASW flow.

D. Incorrect – PK13-01 does not actuate on low ASW flow.

Technical References: AP PK01-01, ASW sys HX Delta P/HDR Press, Revision 21
AP PK01-02, Aux Salt Water PPS Room, Revision 14
AP PK01-03, Aux Salt Water Pumps, Revision 15
LPA-10, Loss of Auxiliary Salt Water, Revision 11
LE-5, Auxiliary Salt Water System, Revision 11

References to be provided to applicants during exam: None

Learning Objective: 5330 – Describe controls, indications, and alarms associated with the ASW System

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	55.41.7, 55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of annunciator alarms, indications, or response procedures: **000065 Loss of Instrument Air / 8**	**Tier #**	1
	Group #	1
	K/A #	2.4.31
	Rating	4.2

Question 15

The plant is at 100% power and the Plant Instrument Air alarm is illuminated. Air Header Pressure indicates 92 psig and is slowly lowering.

Based on these conditions, ARP PK13-16 directs the crew to:

A. Trip the reactor.

B. Verify Reciprocating Air Compressors 0-1 through 0-4 standby start has occurred and that the standby start light (Blue) is lit.

C. Press the Reset button to reset the override start relay.

D. When air header pressure reaches 90 psig, check if letdown has isolated and place excess letdown in service if necessary.

Proposed Answer: B

Explanation:

A. Incorrect. In accordance with ARP PK13-16 if at any time instrument air head drops below 90 psig or is 93 psig and not rising, then go to OP AP-9 "Loss of Instrument Air." The reactor is tripped if there is a loss of control, for instance, the main Feedwater reg valves fail closed.

B. Correct**.** In accordance with ARP PK13-16 section 2.1.2.a "Verify Air Compressors 0-1 through 0-4 standby start has occurred and that the standby start light (Blue) is lit."

C. Incorrect. In accordance with ARP PK13-16, pressing the reset button should be performed per step 2.1.2c "After instrument air header pressure has recovered to normal (100 psig or greater)."

D. Incorrect. In accordance with ARP PK 13-16, various actions are required for Air Header Pressure <90 psig, at 93 psig, from 94-100 psig, and over 100 psig. Letdown will isolate when instrument air to containment is isolated (about 85 psig).

Technical References: ARP PK 13-16

References to be provided to applicants during exam: None

Learning Objective:

Question Source: Bank #

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 41.10 / 45.3	

Examination Outline Cross-Reference	Level	RO
Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	**Tier #**	1
	Group #	1
	K/A #	EA2.2
	Rating	3.4

Question 16

The crew is performing step 6, of EOP ECA-1.1, Loss of Emergency Recirculation, "Determine Containment Spray Requirements"

In order to determine the number of Containment Spray pumps required for containment heat removal, the crew will check containment pressure;

A. Containment Recirculation Sump level, and number of containment fan coolers running.

B. Refueling Water Storage Tank level, and number of containment fan coolers running.

C. Refueling Water Storage Tank level and hydrogen concentration number of containment fan coolers available.

D. Containment Recirculation Sump level, and hydrogen concentration.

Proposed Answer: B

Explanation:

A. Incorrect – Containment Spray pumps do not take a suction off of the Containment Recirculation Sump

B. Correct – EOP ECA-1.1, Step 6 requires RWST level, containment pressure and number of containment coolers running to determine the required number of containment spray pumps.

C. Incorrect – RWST does supply the containment spray pump but, number of required containment spray pumps is affected by the total number of containment fan coolers running not just available.

D. Incorrect - Containment Spray pumps do not take a suction off of the Containment Recirculation Sump, number of required containment spray pumps is affected by the total number of containment fan coolers running not just available.

Technical References: EOP ECA-1.1, Loss of Emergency Recirculation, Revision 24
Background Documents WOG Emergency Response Guidelines, ECA-1.1, Loss of Emergency Coolant Recirculation, Revision 2

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following: **EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.**	**Tier #**	1
	Group #	1
	K/A #	EK2.2
	Rating	3.9

Question 17

GIVEN:

- Unit 1 operators are establishing RCS bleed and feed in accordance with EOP FR-H.1, Loss of Secondary Heat Sink.
- While verifying RCS bleed path per step 18, the Reactor Operator observes that ONE of the Pressurizer PORV's will not open.

Which describes the appropriate action in accordance with EOP FR-H.1, Loss of Secondary Heat Sink, and the reason for this response?

A. Open the Reactor Vessel Head Vent valves because the RCS may not depressurize sufficiently to permit adequate SI flow to remove core decay heat.

B. Close the open PORV and continue efforts to restore AFW flow because two PORVs will not depressurize the RCS sufficiently to allow SI to maintain RCS inventory.

C. No action is required because the RCS will still depressurize sufficiently with two PORVs open to permit adequate SI flow to remove core decay heat.

D. Close the open PORV, then open the Reactor Vessel Head Vent valves to restrict the mass loss sufficiently to ensure RCS inventory can be maintained with SI.

Proposed Answer: A

Explanation:

A. Correct. Open the Reactor Vessel Head Vent valves to allow the RCS to depressurize sufficiently to permit adequate feed of subcooled SI flow to remove core decay heat.

B. Incorrect. The open PORV is to remain open and the RV Head Vent Valves are to be opened toprovide additional RCS depressurization.

C. Incorrect. One PORV may not depressurize the RCS sufficiently to allow enough SI flow tomaintain inventory and remove decay heat.

D. Incorrect. The procedure does not direct the open PORV to be closed. It must remain open to provide RCS depressurization along with the RV head vent valves.

Technical References: EOP FR-H.1, "Loss of Secondary Heat Sink," Background Information for Westinghouse Owners Group Emergency Response Guideline FR-H.1, Operations Lesson: LMCDFRH

References to be provided to applicants during exam:		None

Learning Objective: Explain the plant response to bleed and feed operations. (3817)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.7 / 45.7	

Examination Outline Cross-Reference	Level	RO
Ability to operate and//or monitor the following as they apply to Generator Voltage and Electrical Grid Disturbances: Grid frequency and voltage	**Tier #**	1
	Group #	1
	K/A #	APE 077 AA1.01
	Rating	3.6

Question 18

GIVEN:

- Unit 1 and 2 are operating at 100%
- The series capacitors are NOT in service
- The main generator voltage regulation is in automatic
- A seismic event occurs that causes both Midway lines to be lost

Based on these conditions, grid voltage will ____________ and the generator voltage regulator will _____________ the generator's excitation.

A. Decrease; increase

B. Decrease; decrease

C. Increase; increase

D. Increase; decrease

Proposed Answer: A

Explanation:

A. Correct – Increased line current will result in voltage drop due to E=IR, lower voltage sensed by the voltage regulator will result in an increase in field current which increases the amount of DC current applied to the rotor field windings which control the magnetic field and raises generator terminal voltage.

B. Incorrect - Increased line current will result in voltage drop due to E=IR, decreasing field current would result in a lower generator terminal voltage therefore compounding the event.

C. Incorrect – Voltage will drop due to increases line current E=IR, lower voltage sensed by the voltage regulator will result in an increase in field current which increases the amount of DC current applied to the rotor field windings which control the magnetic field and raises generator terminal voltage

D. Incorrect - Voltage will drop due to increases line current E=IR, decreasing field current would result in a lower generator terminal voltage therefore compounding the even.

Technical References: LJ-4A, Main Generator, Revision 12
LPA-26, Loss of Offsite Power, Revision 7

J4A, Main Generator, Revision 17

References to be provided to applicants during exam: None

Learning Objective: 5280 – Analyze automatic features and interlocks associated with the main generator

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to determine and interpret the following as they apply to the Emergency Boration: **AA2.04 Availability of BWST**	**Tier #**	1
	Group #	2
	K/A #	APE 024 AA2.04
	Rating	3.4

Question 19

GIVEN:

- Unit 1 has tripped from 100% power
- 4 Control Rods are stuck out
- Boric Acid Storage Tank #1 is in service and level is 93%
- Boric Acid Storage Tank #2 level is 91%

Boric Acid Storage Tank #1 level following completion of Emergency Boration will be between _____:

A. 37% and 38%, which meets the minimum required level for plant conditions.

B. 38% and 39%, which does NOT meet the minimum required level for plant conditions.

C. 38% and 39%, which meets the minimum required level for plant conditions.

D. 37% and 38%, which does NOT meet the minimum required level for plant conditions.

Proposed Answer: D

Explanation:

A. Incorrect. This level does not meet the Technical Specification requirement of 14042 gallons.

B. Incorrect. Level will be 3882 gallons, which is between 37% and 38%.

C. Incorrect. Level will be between 37% and 38% and is less than the requirement of 14, 042 gallons.

D. Correct. 4 rods require 900 gallons of boration each. Current level is 7482 gallons – 3600 gallons = 3882 gallons, which is between 37% and 38%. Combine this with Tank #2's 7352 gallons is 11234 gallons, which is less than the minimum required by Technical Specifications of 14042 gallons.

Technical References: Boric Acid Storage Tanks Volume Data, Technical Specifications, OP AP-06 Emergency Boration

References to be provided to applicants during exam: Boric Acid Storage Tanks Volume Data

Learning Objective: Explain the Emergency Boration process (4149)

Question Source: Bank #

(note changes; attach parent)	Modified Bank **#**	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 43.5 / 45.13	

Examination Outline Cross-Reference	Level	RO
Ability to explain and apply system limits and precautions: Related to Loss of Condenser Vacuum	**Tier #**	1
	Group #	2
	K/A #	APE 051 2.1.32
	Rating	3.8

Question 20

Given:

- Unit 1 is currently at 47% power and reducing power to 45%.
- AR-PK10-11, Condenser Pressure/Level is alarming.
- Indicated condenser pressure VB3 is 7.8” Hg and stabilizing.

With these conditions, AP-7, Degraded Condenser, directs the crew to trip the:

A. Turbine, Reactor, AND MFW Pump(s)

B. Turbine AND MFW Pump(s) ONLY

C. Turbine AND Reactor ONLY

D. Turbine ONLY

Proposed Answer: D

Explanation:

A. Incorrect – Reactor trip not required because power is less than the P-9 setting (50%) and the MFW pump trip is at 10” Hg

B. Incorrect - MFW pump trip is at 10” Hg

C. Incorrect - Reactor trip not required because power is less than the P-9 setting (50%)

D. Correct

Technical References: AP-7, Degraded Condenser, Revision 39
AR-PK10-11, Condenser Pressure/Level, Revision 18
LPA-7, Degraded Condenser, Revision 11

References to be provided to applicants during exam: AP-7, Attachment 6.2

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between the Accidental Gaseous Radwaste Release and the following: Auxiliary building ventilation system	**Tier #**	1
	Group #	2
	K/A #	APE 060 AK2.02
	Rating	2.7

Question 21

PK11-25, PLANT VENT RADIATION, and PK11-21, HIGH RADIATION alarm. The source of the radiation is unknown.

The SFM directs operators to enter OP AP-14, Tank Ruptures.

In order to minimize the release, the crew should ______________.

A. Secure the Auxiliary Building Ventilation System to prevent the release of airborne radioactive materials through the plant vent.

B. Place the Auxiliary Building Ventilation System in the BUILDING-and-SAFE GUARDS mode to ensure that the airborne radioactive materials are contained.

C. Place the Auxiliary Building Ventilation System in the SAFE GUARDS only mode with charcoal filters deengergized.

D. Place the Auxiliary Building Ventilation System in the SAFE GUARDS only mode with charcoal filters energized.

Proposed Answer: D

Explanation:

A. Incorrect. The Auxiliary Building Ventilation System is placed into the SAFE GUARDS only mode.

B. Incorrect. While the lineup of the Auxiliary Building Ventilation System will momentarily pass through the BUILDING-and-SAFE GUARDS mode, this is not the desired condition and OP AP-14 directs operators to continue shifting the lineup to SAFE GUARDS only to minimize the release or radiation.

C. Incorrect. The charcoal filters will be energized to reduce relative humidity to less than 70% to improve iodine absorption in the charcoal filters.

D. Correct. Places the Aux Bldg Ventilation in the SAFEGUARDS ONLY Mode, discharging through the Charcoal filters.

References to be provided to applicants during exam: None

Learning Objective: Given an abnormal condition, summarize the major actions of the abnormal operating procedure to mitigate an event in progress. (3477)

Question Source: Bank #

(note changes; attach parent)	Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	CFR 41.7 / 45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reason for the following responses as they apply to High Containment Pressure: Manipulation of controls required to obtain desired operating results during abnormal and emergency situations.	**Tier #**	1
	Group #	2
	K/A #	W/E 14 EK3.3
	Rating	3.5

Question 22

Given:

- The Shift Forman has entered EOP FR-Z.1, Response to High Containment Pressure
- All steam generators are faulted

The minimum feed flow specified in EOP FR-Z.1 for this condition is __________ and the reason for this minimum feed flow is to _________.

Proposed Answer: D

A. 435 gpm; provide sufficient flow to meet heat sink requirements

B. 435 gpm; prevent dryout of the steam generator in order to minimize thermal shock to the steam generator

C. 25 gpm; provide sufficient flow to meet heat sink requirements

D. 25 gpm; prevent dryout of the steam generator in order to minimize thermal shock to the steam generators

Proposed Answer: ___D_

Explanation:

A. Incorrect – EOP E-1, Loss of Reactor or Secondary Coolant requires a minimum feedflow of 435 GPM to an intact steam generator (stem states all steam generators are faulted). WOG BD, E-I, Loss of Reactor or Secondary Coolant, states that the minimum feed flow requirement satisfies the feed flow requirement of the Heat Sink Status Tree until level in at least one SG is restored into the narrow range.

B. Incorrect – EOP E-1, Loss of Reactor or Secondary Coolant requires a minimum feedflow of 435 GPM to intact steam generator (stem states all steam generators are faulted). WOG BD, FR-Z.1, Response to High Containment Pressure, states that 25 GPM maintains the steam generator in a wet condition, thereby minimizing any thermal shock effects if feed flow is increased.

C. Incorrect – EOP FR-Z.1, Caution 2, states, if all steam generators are faulted, at least 25 GPM should be maintained to each steam generator. EOP E-1, Loss of Reactor or Secondary Coolant, WOG BD, FR-Z.1, Response to High Containment Pressure, states that 25 GPM maintains the steam generator in a wet condition, thereby minimizing any thermal shock effects if feed flow is increased.

D. Correct – EOP FR-Z.1, Caution 2, states, if all steam generators are faulted, at least 25 GPM should be maintained to each steam generator. WOG BD, FR-Z.1, Response to High Containment Pressure, states that 25 GPM maintains the steam generator in a wet condition, thereby minimizing any thermal shock effects if feed flow is increased.

Technical References: EOP E-1, Loss of Reactor or Secondary Coolant, Revision 25A
EOP FR-Z.1, Response to High Containment Pressure, Revision 10
WOG BD, E-1, Loss of Reactor or Secondary Coolant, Revision 2
WOG BD, FR-Z.1, Response to High Containment Pressure, Revision 2

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/**Fundamental**	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5, 55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: Failed fuel-monitoring equipment	**Tier #**	1
	Group #	2
	K/A #	APE 076 AA1.04
	Rating	3.2

Question 23

Initial Conditions:

- Unit 1 tripped from 100% power 10 hours ago. Core Exit Thermocouples have been indicating higher than expected temperatures.
- The Post Accident Sampling System is inoperable.
- You are directed by the SFM to place Letdown on service.

If the fuel has suffered cladding damage, high reactor coolant activity from the failed fuel will:

A. be cleaned up by the ion exchanger and remain unnoticed until the PASS is restored and reactor coolant is sampled.

B. cause area radiation monitors in the Auxiliary Building indicate higher in the areas of CVCS and seal injection.

C. will cause radiation monitors in the Auxiliary Building indicate higher in the area of the steam generator blowdown valves.

D. only affect radiation levels in the Auxiliary Building if there is a Letdown Line Failure.

Proposed Answer: B

Explanation:

A. Incorrect. The Ion Exchanger will not remove all of the activity from the Reactor Coolant.

B. Correct. Radiation levels will increase at the Letdown Heat Exchanger and the RCP seal injection.

C. Incorrect. With no steam generator tube rupture, there is no reason for radiation levels at secondary valves to increase.

D. Incorrect. While a Letdown Line Failure will cause radiation levels in the Auxiliary Building to increase, that is not the ONLY reason for increased radiation levels.

Technical References: Operations Lesson LMCDCDA Core Damage Assessment; Operations Lesson Chemical and Volume Control System

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the CVCS. (40449)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	CFR 41.7 / 45.5 / 45.6	

<table>
<tr><td>Examination Outline Cross-Reference</td><td>Level</td><td>RO</td></tr>
<tr><td rowspan="5">Knowledge of the specific basis for EOPs: Re-diagnosis and SI termination</td><td>Tier #</td><td>1</td></tr>
<tr><td>Group #</td><td>2</td></tr>
<tr><td>K/A #</td><td>W/E01
2.4.18</td></tr>
<tr><td>Rating</td><td>3.3</td></tr>
</table>

Question 24

Unit 1 operators are performing EOP E-1.1, SI Termination, following a LOCA.

According to procedure EOP E-1.1, the preferred RCP to run is:

A. RCP 2 because it provides normal pressurizer spray capabilities.

B. RCP 2 because it reduces the pressurizer level and pressure transients.

C. RCP 1 because it provides normal pressurizer spray capabilities.

D. RCP 1 because reduces the pressurizer level and pressure transients.

Proposed Answer: A.

Explanation:

A. Correct – Step 24 of EOP E-1.1 states to verify RCP 2 running for spray capabilities.

B. Incorrect – reduced pressurizer level and pressure transients is due maintaining a saturated system in the pressurizer.

C. Incorrect – Step 24 of EOP E-1.1 states that RCP 2 is checked operating to obtain normal pressurizer spray capabilities and if RCP 2 is not available then more than one running RCP may be necessary to provide normal pressurizer spray capabilities.

D. Incorrect - reduced pressurizer level and pressure transients is due maintaining a saturated system in the pressurizer.

Technical References: LPE-1B, E-1.1, SI Termination, Revision 12
EOP E-1.1, SI Termination, Revision 26
WOGBD ES-1.1, SI Termination, Revision 2

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between the (Containment Flooding) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	**Tier #**	1
	Group #	2
	K/A #	WE15 EK2.2
	Rating	2.7

Question 25

Unit 1 has experienced a large break Loss of Coolant Accident. The SFM has entered EOP FR-Z.2, "Response to Containment Flooding," due to the Critical Safety Function Status Tree, F 0.5, CONTAINMENT, being MAGENTA. The Containment Recirc Sump level is 95 feet.

According to this procedure, the minimum level the containment would be considered to be flooded is a Containment Recirc Sump level greater than _______ feet.

A. 95

B. 95.75

C. 96

D. 97.5

Proposed Answer: B

Explanation:

A. Incorrect. FRZ-2 step 1 directs operators to exit Flooding in Containment for recirc sump levels less than 95.75 feet. The Response to Containment Flooding Background Document, FRZ-2, states, "The maximum level of water in the containment following a major accident generally is based upon the entire water contents of the reactor coolant system, refueling water storage tank, condensate storage tank, and SI accumulators."

B. Correct. This is the water volume from the RCS, RWST, CST, and 4 SI Accumulators. This approximates the maximum water volume introduced following a LOCA plus a steamline or feedline break inside of containment.

C. Incorrect. The correct level per step 1 of EOP FRZ-2 is 95.75 feet. The maximum level of water in the containment following a major accident generally is based upon the entire water contents of the reactor coolant system, refueling water storage tank, condensate storage tank, and SI accumulators.

D. Incorrect. The correct level per step 1 of EOP FRZ-2 is 95.75 feet.

Technical References: EOP F-0, "Critical Safety Function Status Trees," EOP FR-Z.2, "Response to Containment Flooding," Operations Lesson Mitigating Core Damage – Containment, Systems Lesson Containment Structure.

Response to Containment Flooding Background Document, FRZ-2,

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the Containment Structure. (37590)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 41.7 / 45.7	

Examination Outline Cross-Reference	Level	RO
Ability to operate and / or monitor the following as they apply to the Natural Circulation Cooldown: Operating behavior characteristics of the facility	**Tier #**	1
	Group #	2
	K/A #	EA1.2
	Rating	3.6

Question 26

GIVEN:

Unit 2 is performing a Natural Circulation Cooldown in accordance with EOP E-0.2, Natural Circulation Cooldown, with only one CRDM fan running.

Given these conditions, the maximum cooldown rate that the crew can establish is _____ degrees/hr.

A. 25

B. 50

C. 90 (Check on Admin CD limit)

D. 100

Proposed Answer: B

Explanation:

A. Incorrect – Less than 25°F/hr cooldown is the maximum rate for Unit 1 in EOP E-0.2.

B. Correct – Unit 2 EOP E-0.2, Step 10 states a cooldown rate of 50°F/hr

C. Incorrect – A 90°F/hr cooldown rate is the administrative limit for a normal cooldown.

D. Incorrect – A 100°F/hr cooldown rate is the tech spec limit for a normal cooldown.

Technical References: EOP E0.2, U1, Natural Circulation Cooldown, Revision 23
EOP E-0.2, U2, Natural Circulation Cooldown, Revision 18
Operational Phase Training LPE-0.2, Natural Circulation Cooldown, Revision 14

References to be provided to applicants during exam: None

Learning Objective:

Question Source: Bank #
(note changes; attach parent) Modified Bank #
New X

Question History: Last NRC Exam NA

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Ability to determine and interpret the following as they apply to the (Pressurized Thermal Shock): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	**Tier #**	1
	Group #	2
	K/A #	WE08 EA2.1
	Rating	3.4

Question 27

Initial conditions on Unit 1:

- RCS T_{COLD} is 170°F and lowering slowly
- All T_{COLDs} have decreased approximately 110°F in the last hour
- Reactor Coolant System Pressure is 33 psig and slowly lowering
- Containment Pressure is 25 psig and slowly lowering
- RVLIS indicates 15 feet above the Core Plate

The Shift Foreman directs operators to enter EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, due to exceeding the maximum RCS cooldown rate of 100 degrees per hour. RCS pressure/temperature being to the Left of the Limit A curve.

The first major step of this procedure is to:

A. Initiate Auxiliary Spray.

B. Terminate Safety Injection if the criteria is satisfied.

C. Reduce RCS cooldown by throttling ECCS flow.

D. Verify RCS pressure is less than RHR Pump shutoff head.

Proposed Answer: D

Explanation:

A. Incorrect. Initiating Auxiliary Spray would be correct for entering FR-P.2 Anticipated Pressurized Thermal Shock Condition or for later steps in EOP FR-P.1.

B. Incorrect. This occurs early in EOP FR-P.1, but not until step 6 and after D. It does minimize cool down of the cold leg.

C. Incorrect. ECCS flow has caused the cooldown but conditions are not met to reduce ECCS flow.

D. Correct. The action is to determine if a large break Loss of Coolant Accident exists because if so, then Pressurized Thermal Shock is not a serious concern.

Technical References: EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition; EOP FR-P.2 Response to Anticipated Pressurized Thermal

Shock Condition; EOP F-0 Critical Safety Function Status Trees part F-0.4 RCS Integrity; Operations Lesson RCS Integrity FRs

References to be provided to applicants during exam: None

Learning Objective: Identify entry conditions for the FRPs. (9704)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental **Comprehensive/Analysis**	 X
10CFR Part 55 Content:	CFR: 43.10 / 45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Containment isolation valves affecting RCP operation	**Tier #**	2
	Group #	1
	K/A #	003 K6.04
	Rating	2.8

Question 28

Unit 1 is at 100% and a Phase A containment isolation occurs.

The Phase A containment isolation signal will immediately close:

A. CVCS-8112, RCPs #1 Seal Outlet valve, diverting flow through RV-8121 to the PRT

B. FCV-355, CCW Header C Supply valve, isolating CCW flow to the RCPs

C. FCV-363, RCP Lube Oil Cooler Return valve, isolating CCW to the RCP lube oil cooler

D. CVCS-8141A/B/C/D, RCP Seal Leakoff valves, isolating Seal 1 leakoff from the RCPs

Proposed Answer: A.

Explanation:

A. Correct – Phase 'A' Containment Isolation closes CVCS-8100 and CVCS-8112, RCPs #1 Seal Outlet valve. This results in RV-8121 lifting and diverting flow to the PRT

B. Incorrect – FCV-355 is isolated by a Phase 'B' Containment Isolation signal

C. Incorrect – FCV-363 is isolated by a Phase 'B' Containment Isolation signal

D. Incorrect – CVCS-8141A/B/C/D fail open on a loss of instrument air following a Phase 'A' Containment Isolation signal.

Technical References: OP AP-9, Loss of Instrument Air, Revision
STG B-6A, Reactor Protection System Supplement, Revision 1
STG B-6A, Protection Systems, Revision 17
STG B-1A, Chemical and Volume Control System, Revision 18
STG F-2, Component Cooling Water, Revision 18

References to be provided to applicants during exam: None

Learning Objective:

Question Source: Bank #
(note changes; attach parent) Modified Bank #
New X

Question History: Last NRC Exam NA

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls including: Activity levels in primary system	**Tier #**	2
	Group #	1
	K/A #	004 A1.01
	Rating	2.9

Question 29

GIVEN:

- Unit 1 is at 100% power
- Volume Control Tank (VCT) Level is decreasing
- Pressurizer Level and Reactor Coolant System (RCS) Pressure are decreasing
- Letdown flow is abnormally low
- Letdown Heat Exchanger Room is 155°F
- PK 4-21, LETDOWN PRESS/FLO/TEMP, alarm is in
- Containment Sump Level is increasing

Based on the continued operation of the Chemical and Volume Control System , the operators should:

A. Place Excess Letdown in service and carefully monitor RCS chemistry and activity.

B. Place Excess Letdown in service and begin a cooldown of the RCS.

C. Start another Coolant Charging Pump

D. Pump the Containment Sump to the Floor Drain Receivers.

Proposed Answer: A

*Note: 8149A/B/C closed auto on LTDN H/X room . 150F

Explanation:

A. Correct. A Letdown line rupture has occurred downstream of CVCS-1-8152 as evidenced by the High Letdown Heat Exchanger room temperature and initial conditions. Also, Letdown Isolation valves 8149A/B/C failed shut when Letdown Heat Exchanger room temperature reached 150°F. OP AP-18, Letdown Line Failure, directs operators to place Excess Letdown on service and OP B-1A IV, CVCS Excess Letdown Place in Service, cautions operators, “When the normal letdown flowpath is out of service and excess letdown is the only letdown flowpath, RCS chemistry and activity should be carefully monitored since the CVCS demins, filters and volume control tank are bypassed during this mode.”

B. Incorrect. All operations that affect RCS Inventory Balance should be suspended, including heatups and cooldowns.

C. Incorrect. Symptoms for Loss of Charging are similar, however, the lowering VCT level differentiates between the two casualties; actions per OP AP-17, Loss of Charging are incorrect. If Letdown is not operating properly, increasing Charging will not lead to proper resolution of the malfunction.

D. Incorrect. OP AP-17, Loss of Charging actions are not correct. Pumping the Containment Sump to the Floor Drain Receivers might be beneficial but does not address trends, initial conditions or system operating conditions associated with alarm PK4-21 LETDOWN PRESS/FLO/TEMP.

Technical References: OP AP-17, Loss of Charging; OP AP-18, Letdown Line Failure; OP B-1A IV, CVCS Excess Letdown Place in Service; PK4-21 LETDOWN PRESS/FLO/TEMP; System Training Guide Chemical and Volume Control System, B-1A; Operations Lesson Chemical and Volume Control System, LB-1A.

References to be provided to applicants during exam: None

Learning Objective: Discuss significant precautions and limitations associated with the CVCS. (5093)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.5 / 45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect of a loss or malfunction on the RHRS will have on the following: ECCS	**Tier #**	2
	Group #	1
	K/A #	005 K3.05
	Rating	3.7

Question 30

Unit 1 is aligned for hot leg recirculation in accordance with EOP E-1.4, Hot Leg Recirculation following a LOCA event. Actions for EOP E-1, Loss of Reactor or Secondary Coolant are in progress.

If SI-8982A, Containment Sump to RHR Pump Suction, were to inadvertently close, the effect on the Emergency Core Cooling System would be all flow to:

A. ONLY the ECCS CCPs would be secured.

B. BOTH the ECCS CCPs AND SIP 1-1 would be secured.

C. BOTH the ECCS CCPs AND SIP 1-2 would be secured.

D. ONLY the SIPs would be secured.

Proposed Answer: C

Explanation:

A. Incorrect – SI-8982A supplies RHR Pump ‘A’ which provides flow to the suction of the CCPs in addition to SIP 1-2.

B. Incorrect - SI-8982A supplies RHR Pump ‘A’ which provides flow to the suction of the CCPs in addition to SIP 1-2. RHR Pump ‘B’ supplies flow to the suction of SIP 1-1.

C. Correct - SI-8982A supplies RHR Pump ‘A’ which provides flow to the suction of the CCPs in addition to SIP 1-2.

D. Incorrect - SI-8982A supplies RHR Pump ‘A’ which only provides flow to the suction of SIP 1-2.

Technical References: System Training Guide B3, Emergency Core Cooling System, Revision 19, Operational Phase Lesson Plan LPE-1C, Recirculation Modes and LOCA Outside Containment, Revision 11

References to be provided to applicants during exam: None

Learning Objective:

Question Source: Bank #
(note changes; attach parent) Modified Bank #
New X

Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operation of the ECCS, including: Pumps	**Tier #**	2
	Group #	1
	K/A #	006 A3.02
	Rating	4.1

Question 31

Unit 1 plant conditions:

- A Loss of Coolant Accident (LOCA) has occurred
- Reactor Coolant Pressure is 1510 psig
- Containment Pressure is 27 psig
- Safety Injection (SI) Train A failed to actuate
- PK02-02, SAFETY INJECTION INITIATE, is alarming
- Level of the Refueling Water Storage Tank is 32%
- EOP E-0, Reactor Trip or Safety Injection, Appendix E, is the procedure in effect

The status of Emergency Core Cooling System Components is:

1. SI Pump(s) total flow is 80 gpm
2. Charging Flow rate is approximately100 gpm
3. RHR pumps have tripped
4. Low Head Injection is recirculating through 641B

A. 1 and 2

B. 1 and 3

C. 2 and 3

D. 1 and 4

Proposed Answer: B

Explanation:

A. Incorrect. Charging flow at 1510 psig is approximately 400 gpm per charging pump.

B. Correct. Only the train B Safety Injection pump is operating; at 1510 psig the flow rate will be approximately 80 gpm. When the RWST reaches 33%, the RHR pumps trip off and must be manually aligned for cold leg recirculation. The remaining water in the RWST will be used by the Containment Spray pumps.

C. Incorrect. Charging flow at 1510 psig is approximately 400 gpm per charging pump.

D. Incorrect. Low Injection uses the RHR pumps, which trip off when RWST level reaches 33%.

Technical References: System Lesson Emergency Core Coolings System, Operations Lesson

Emergency Core Cooling System, EOP E-0 Reactor Trip or Safety Injection

References to be provided to applicants during exam:		None

Learning Objective: Analyze automatic features and interlocks associated with the Emergency Core Cooling System. (8045)

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.7 / 45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and / or cause effect relationships between the PRTS and the following systems: RCS	**Tier #**	2
	Group #	1
	K/A #	007 K1.03
	Rating	3.0

Question 32

GIVEN:

- Unit 1 experienced a faulted steam generator outside containment resulting in a Reactor Trip and Safety Injection
- The faulted steam generator has been isolated
- RCS pressure is 1800 psig and increasing
- PK05-25, PRT Pressure Hi/Low, is alarming
- PRT level and pressure are RISING slowly

The reason the PRT parameters are rising is:

A. A Pressurizer Power Operated Relief Valve is open.

B. Reactor Coolant Pump Seal Water Return flow is going to the PRT.

C. Normal letdown flow is diverted to the PRT through RV-8117, Letdown Relief Valve.

D. RCS-8030, Primary Water valve to the PRT, has failed open due to instrument air isolation to containment.

Proposed Answer: B

Explanation:

A. Incorrect – The PORVs will discharge to the PRT but RCS pressure is 1800 psig below PORV lift setting and RCS pressure is increasing.

B. Correct – Seal Water Return is isolated by the Safety Injection Signal which diverts water through S121, Seal Water Return Relief Valve to the PRT.

C. Incorrect – Letdown was isolated by the Safety Injection Signal. RV-8117 will not lift to relieve to the PRT because letdown is isolated.

D. Incorrect – RCS-8030 is isolated from the primary water supply by RCS-8029 following Phase 'A' Containment Isolation preventing water to flow to the PRT.

Technical References: STG A-4B, Pressurizer Relief Tank, Revision 13
STG B-1A, Chemical and Volume Control System, Revision 18
STG A-4A, Pressurizer, Pressure and Level Control, Revision 17
AR PK05-25, PRT Press/LVL Temp, Revision 15

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.2-9	

Examination Outline Cross-Reference	Level	RO
Ability to locate and operate components, including local controls. 008 Component Cooling Water	Tier #	2
	Group #	1
	K/A #	008 2.1.30
	Rating	4.4

Question 33

Unit 1 is in MODE 4 preparing to enter MODE 3.

Alarm AR PK01-06, CCW Vital Header A/B, alarm comes in and the control room finds that the alarm input is number 459, RHR HX 1-2 CCW FLO Hi-Lo.

The Aux Watch has been tasked to verify Component Cooling Water (CCW) flow on the outlet of the affected RHR HX. This would be performed by checking CCW flow on the outlet of the RHR HX:

A. In number 2 room at the 115 foot elevation of the Auxiliary Building. . If flow through the RHR HX is low, then throttle the RHR HX Number 2 CCW outlet valve, CCW-1-151.

B. At the 85 foot elevation of the turbine building. If flow through the RHR HX is low, then open Containment Fan Cooling Unit (CFCU) 1-3 outlet valve, CCW-1-476.

C. At the 100 foot elevation of the GE-GW Penetration passageway. If flow through the RHR HX is low, then throttle the RHR HX Number 2 CCW outlet valve, CCW-1-151.

D. At the 100 foot elevation of the GE-GW Penetration passageway. If flow through the RHR HX is low, then throttle the RHR HX Number 2 CCW outlet valve, CCW-1-151.

Proposed Answer: C

Explanation:

A. Incorrect. The RHR #2 HX is at this location, but not the outlet valves for CCW.

B. Incorrect. The CCW HXs are located at the 85 foot elevation of the turbine building. Also operating CVCU 1-3 outlet valve is the action to take if flow is too high.

C. Correct. The correct location to observe RHR HX CCW flow is the 100 foot elevation of the GE-GW Penetration passageway. If flow is too low, the correct action is to throttle the RHR HX Number 2 CCW outlet valve, CCW-1-151.

D. Incorrect. Throttling flow to RHR HX Number 1 CCW outlet valve would not correct the given condition since the alarm came in on RHR HX 1-2 (Unit 1, heat exchanger number 2).

Technical References: System Lesson Component Cooling Water, Operations Lesson Component Cooling Water, AR PK 01-06.

References to be provided to applicants during exam:		None

Learning Objective: Identify the location of components associated with the CCW System. (8128)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 41.7 / 45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the bus power supplies to the following: Controller for PZR spray valve	**Tier #**	2
	Group #	1
	K/A #	K2.02
	Rating	2.5

Question 34

GIVEN:

The pressurizer spray valves would fail closed if 125 VDC __________ was lost.

A. Bus 11

B. Bus 12

C. Bus 13

D. Bus 15

Proposed Answer: C

Explanation:

A. Incorrect – Loss of 125 VDC Bus 11 does not affect the spray valves.

B. Incorrect – Loss of 125 VDC Bus 12 does not affect the spray valves.

C. Correct – The loss of 125 VDC vital bus 13 results in FCV-584, Containment Instrument Air Supply Valve, to fail closed resulting in the depressurization of the instrument air header inside containment. Both spray valves are air opened and fail-closed valves, therefore both valves will close following a loss of 125 VDC vital bus 13.

D. Incorrect – Loss of 125 VDC Bus 15 does not affect the spray valves and is a non-vital supply.

Technical References: AP-23 - Loss of Vital DC Bus, Revision 13

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X

10CFR Part 55 Content: 55.41.7

Examination Outline Cross-Reference	Level	RO
Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: Automatic or manual enable/disable of RPS trips	Tier #	2
	Group #	1
	K/A #	012 K4 06
	Rating	3.2

Question 35

Unit 1 is performing a reactor and secondary plant startup. Currently;

- The main generator output is 140 MWe
- Power Range Channel A is in Bypass for calibration
- Pressurizer Low Pressure Channel A is in Bypass for surveillance testing
- Reactor Coolant Pump (RCP) #3 breaker tripped open
- Pressurizer Low Pressure Channel B now fails low

Based on these conditions:

1. Pressurizer Low Pressure Trip is enabled but the reactor will not trip.
2. The Power Range High Flux Low Trip is automatically blocked.
3. Pressurizer Low Pressure Trip must be manually enabled and will result in a reactor trip.
4. RCS Low Flow trip is automatically enabled but the reactor will not trip.

A. 1 and 4

B. 1 and 2

C. 2 and 3

D. 2 and 4

Proposed Answer: A

Explanation:

A. Correct. 140 MWe is about 18-20% reactor power, which energizes P-10, Power Range at Power Permissive (2 of 4 NI's >10%). Turbine first stage pressure above 10% power deenergizes P-13, Turbine Low Power Permissive. P-10 (energized) and P-13 (deenergized) are both inputs to P-7, Low Power Permissive. Therefore, Pressurize Low Pressure Trip from P-7 is automatically enabled and at this reactor power level, RCS Pressure is >1950 psig - the reactor will not trip. The RCS Low Flow trip is also automatically enabled by P-7, but will not result in a trip because the coincidence logic requires 2 out of 4 loops to have a low flow condition until reactor power is above P-8, Loss of Flow Permissive setpoint of 35% reactor power.

B. Incorrect. When P-10 is energized, it allows the Power Range High Flux Low Trip to be manually blocked.

C. Incorrect. When P-10 is energized, it allows the Power Range High Flux Low Trip to be manually blocked. When P-7 is energized, the Pressurizer Low Pressure Trip is automatically enabled.

D. Incorrect. When P-10 is energized, it allows the Power Range High Flux Low Trip to be manually blocked.

Technical References: OP L-3, Secondary Plant Startup; OP C-3.II, Main Unit Turbine Startup; Operations Lesson Reactor Protection System, System Lesson Reactor Protection System

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Reactor Protection System. (37048)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.7	

Examination Outline Cross-Reference	Level	RO
Ability to manually operate and/or monitor in the control room: ESFAS initiated equipment which fails to actuate	**Tier #**	2
	Group #	1
	K/A #	013 A4.01
	Rating	4.5

Question 36

GIVEN:

- Unit 1 has tripped
- Alarm PK04-14, Reactor Trip Actuated, – On
- Alarm PK08-21, Safety Injection Actuated, – On
- Feedwater Isolation portion of Monitor Light Box C
  - Red Activated Lights are off
  - White Status Lights are on
- S/G Level portion of Monitor Light Box C
  - Red Activated Lights are on
  - White Status Lights are off

EOP E-0, Reactor Trip or Safety Injection, is in use.

According to EOP E-0, the operators should manually close the S/G Main Feedwater_________:

A. Isolation Valves with White Status Lights – On ONLY

B. Isolation AND Control Bypass Valves with White Status Lights - On

C. Isolation AND Control Valves with White Status Lights - On

D. Control AND Control Bypass Valves with White Status Lights - On

Proposed Answer: D

Explanation:

A. Incorrect – Action required if S/G Level portion of Monitor Light Box C Red Activated Lights – On and White Status Lights – Off are not met.

B. Incorrect – Feedwater Isolation Valves are indicated on the S/G Level portion of the Monitor Light Box C and Control Bypass Valves are indicated on the Feedwater Isolation portion of Monitor Light Box C.

C. Incorrect - Feedwater Isolation Valves are indicated on the S/G Level portion of the Monitor Light Box C and Control Valves are indicated on the Feedwater Isolation portion of Monitor Light Box C.

D. Correct – S/G Main Feedwater Control and Control Bypass Valves are indicated on Monitor Box C.

Technical References: EOP E-0, Reactor Trip or Safety Injection, Revision 40
LPE-0, Reactor Trip and Safety Injection Response, Revision 11

References to be provided to applicants during exam: None

Learning Objective: 3798 - Explain the means to verify ECCS injection valve alignment

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of service water	**Tier #**	2
	Group #	1
	K/A #	022 A2.04
	Rating	2.9

Question 37

A Steam Line Rupture has occurred in Unit 1.

GIVEN:

- RCS Pressure is 1712 psig and slowly rising
- Containment Pressure is 29 psig and slowly rising
- All Emergency Core Cooling Systems and Safety Injection systems actuated as designed
- Auxiliary Saltwater (ASW) Pumps #1 and #2 have tripped and cannot be restarted
- The Refueling Water Storage Tank (RWST) level is 3%
- Unit 2 is at 100% power

In order to continue the cooldown of Unit 1;

A. The RHR pumps tripped when the RWST reached 33% and must be aligned to take a suction from the Containment Structure Sump.

B. The RHR pumps tripped when the RWST reached 33% and must be aligned for cold leg injection. The Containment Spray pumps will use water from the Boric Acid Storage Tank (BAST) to continue providing flow to the CS system.

C. Component Cooling Water (CCW) has lost cooling water. In accordance with OP AP-11, Malfunction of CCW System, Appendix D, Loss of the Ultimate Heat Sink, line up fire water to cool the CCW heat exchangers.

D. The Containment Spray (CS) System is in danger of being lost. Per OP AP-10, Loss of Auxiliary Saltwater, open the Unit 1 to Unit 2 ASW cross-tie valve, FCV-601, to restore ASW cooling.

Proposed Answer: D

Explanation:

A. Incorrect. After tripping off when the RWST level reached 33%, the RHR pumps must be manually aligned to take a suction from the Containment Recirculation Sump.

B. Incorrect. While the RHR pumps do trip off at 33% level in the RWST and must be manually aligned for cold leg injection, however, the Containment Spray Pumps will provide forced flow for the CS system from the RWST, not the BAST.

C. Incorrect. CCW cooling has been lost, as has cooling to all of CCW's loads. Loss of both

ASW pumps puts the crew in OP AP-10 Loss of Auxiliary Saltwater, not OP AP-11 Malfunction of Component Cooling Water. Additionally, the site was designed so that ASW could be cross-connected between Unit 1 and Unit 2 (ie Unit 2 may supply ASW to Unit 1 and vice versa).

D. **Correct.** The RHR heat exchangers are cooled by CCW, and the CCW heat exchangers are cooled by ASW. OP AP-10 directs operators to open the Unit 1 and Unit 2 ASW cross-tie valve.

Technical References: OP AP-10, Loss of Auxiliary Saltwater; OP AP-11, Malfunction of Component Cooling Water System including Appendix D, Loss of the Ultimate Heat Sink; systems lesson Containment Spray; Operations Lesson Emergency Core Cooling Systems.

References to be provided to applicants during exam: None

Learning Objective: Describe system interrelationships between the ASW System and other plant systems. (3785)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.5 / 43.5 / 45.3 / 45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of operational implications of the following concepts as they apply to the Hydrogen Recombiner and Purge System and flammable hydrogen concentration.	**Tier #**	2
	Group #	1
	K/A #	028 K5.02
	Rating	3.4

Question 38

The Hydrogen Purge System is placed in service at _____ hydrogen concentration and the maximum allowed hydrogen concentration for using the recombiners is ______.

A. 3.5%; 3.5%

B. 3.5%; 4.0%

C. 4.0%; 4.0%

D. 4.0%; 5%

Proposed Answer: B

Explanation:

A. Incorrect – 3.5% is correct for placing the Hydrogen Purge system in Service in accordance with OP H-8:I. The limit for operation of the Hydrogen Recombiners in accordance with OP H-9 is 4.0% not 3.5%.

B. Correct - 3.5% is correct for placing the Hydrogen Purge system in Service in accordance with OP H-8:I. The limit for operation of the Hydrogen Recombiners in accordance with OP H-9 is 4.0%.

C. Incorrect - 3.5% not 4.0% is correct for placing the Hydrogen Purge system in Service in accordance with OP H-8:I. The limit for operation of the Hydrogen Recombiners in accordance with OP H-9 is 4.0%.

D. Incorrect – 3.5% not 4.0% is correct for placing the Hydrogen Purge system in Service in accordance with OP H-8:I. The limit for operation of the Hydrogen Recombiners in accordance with OP H-9 is 4.0% not 3.5%

Technical References: STG LH-8, Containment Hydrogen Purge System, Revision 11
STG LH-9, Containment Hydrogen Recombiners, Revision 11A
OP H-8:I, Containment Hydrogen Purge System – Make Available and Place in Service, Revision 8
OP H-9, Inside Containment H2 Recombination System, Revision 10

References to be provided to applicants during exam: None

Learning Objective: Discuss significant precautions and limitations associated with the Containment Hydrogen Recombiner System
3863 - Describe the operation of the Containment Hydrogen Purge System

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment pressure	**Tier #**	2
	Group #	1
	K/A #	026 A1.01
	Rating	3.9

Question 39

GIVEN:

- A LOCA has occurred on Unit 1
- Train A of Containment Spray is inoperable
- Safety Injection has actuated
- Containment Pressure is 23 psig and rising
- Only 2 Containment Fan Cooling Units (CFCUs) are operating
- The crew is about to transition from E-0, Reactor Trip or Safety Injection

Based on these conditions, Containment Spray Train B __________ actuated and containment pressure ____________.

A. has not; will exceed 57 psig.

B. has; will NOT exceed 47 psig.

C. has; will exceed 47 psig.

D. has not; will NOT exceed 57 psig.

Proposed Answer: B

Explanation:

A. Incorrect. CS Train B will have actuated on Hi Hi Containment Pressure at 22 psig. The design limit for Containment Pressure is 47 psig; pressure will not exceed 57 psig.

B. Correct. CS Train B will have actuated on Hi Hi Containment Pressure at 22 psig. One train of CS and two CFCUs in operation will prevent Containment Pressure from exceeding the design limit of 47 psig.

C. Incorrect. Containment Pressure will NOT exceed 47 psig.

D. Incorrect. CS Train B will have actuated on Hi Hi Containment Pressure at 22 psig. Containment Pressure will not exceed 47 psig.

Technical References: Operations Lesson Reactor Protection System, Operations Lesson Containment Spray System, Final Safety Analysis Report section 6.2

References to be provided to applicants during exam: None

Learning Objective: Explain significant Containment Spray System design features and the importance to nuclear safety. (40802)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	CFR: 41.5 / 45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause effect relationships between the MRSS and the following systems: AFW	**Tier #**	2
	Group #	1
	K/A #	039 K1.07
	Rating	3.4

Question 40

The TDAFW Pump steam supply is from steam generators:

A. 1-1 and 1-2 upstream of the Main Steam Isolation Valves.

B. 1-1 and 1-2 downstream of the Main Steam Isolation Valves.

C. 1-2 and 1-3 upstream of the Main Steam Isolation Valves.

D. 1-2 and 1-3 downstream of the Main Steam Isolation Valves.

Proposed Answer: C

Explanation:

A. Incorrect – Supplied from SG 2 and 3.

B. Incorrect – Supplied from SG 2 and 3 and upstream of the MSIVs.

C. Correct.

D. Incorrect – Supplied from upstream of the MSIVs.

Technical References: STG C-2A, Main Steam System, Revision 14

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.2-9	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operation of the MFW, including: Feedwater isolation	**Tier #**	2
	Group #	1
	K/A #	059 A3.06
	Rating	3.2

Question 41

GIVEN:

- Unit 1 is at 100% power
- Reactor Coolant System (RCS) pressure has begun to lower steadily
- Steam Generator (SG) 1-1 Water Level is 90% Narrow Range and increasing rapidly
- AR PK09-01, SG 1-1 Press, Lvl Flow alarm comes in
- AR PK12-11, Turbine Trip, now comes in

Given these conditions, what is the status of the Main Feedwater System?

1. SSPS Train A has tripped the MFW Pump Turbines
2. SSPS Train B automatically shut the Main Feed Isolation Valves (FCV-438, FCV-439, FCV-440, and FCV-441)
3. Steam Generator blowdown and sample valves automatically isolated
4. Main Feed Reg Bypass Valves (FCV-1510, FCV-1520, FCV-1530, and FCV-1540) automatically isolated

A. 1 and 2

B. 2 and 3

C. 3 and 4

D. 1 and 4

Proposed Answer: C

Explanation:

A. Incorrect. 1: SSPS Train B trips the MFW Pumps, not SSPS Train A. 2: SSPS Train A shuts the MFIVs, not SSPS Train B.

B. Incorrect. 2: SSPS Train A shuts the MFIVs, not SSPS Train B. 3: Correct, the start of any AFW pump isolates SG blowdown and sample valves.

C. Correct. 3: Correct, the start of any AFW pump isolates SG blowdown and sample valves. 4: SGWL at 90% energizes P-14, which shuts the Main Feed Reg Bypass Valves.

D. Incorrect. 1: SSPS Train B trips the MFW Pumps, not SSPS Train A.

Technical References: Operations lessons Main Feedwater System and Reactor Protection System; System Lessons Main Feedwater System and Auxiliary Feedwater System; Reactor Protection System Supplement

References to be provided to applicants during exam: None

Learning Objective: Describe system interrelationships between the Main Feedwater System and other plant systems. (37615)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.7 / 45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the following concepts as they apply to AFW: Relationship between AFW flow and RCS heat transfer	**Tier #**	2
	Group #	1
	K/A #	K5.01
	Rating	3.6

Question 42

The basis for Unit 2 maintaining an AFW flowrate of at least 440 GPM while performing a plant shutdown from an extended full power run is to:

A. Provide for a 50°F/Hr RCS cooldown rate.

B. Maintain sufficient steam generator inventory to prevent lifting a relief valve due to RCS heatup.

C. Restore normal steam generator water level within 2 hours in order to provide for decay heat removal.

D. Provide for a 100°F/Hr RCS cooldown rate.

Proposed Answer: B

Explanation:

A. Incorrect – UFSAR Section 6.5 states that the flowrate for the 50°F/Hr RCS cooldown rate following a plant shutdown from the maximum calculated NSSS output of 3568 MWt occurs at about 52 seconds after the shutdown with all three AFW pump operating.

B. Correct – UFSAR Section 6.5 states that net flowrate of 440 GPM is sufficient to prevent the water in the steam generators from reaching less than the minimum level required to prevent heatup of the RCS to the point where water relief would occur.

C. Incorrect – Normal steam generator water level is restored in approximately 3.2 hours not 2 hours in accordance with UFSAR Section 6.5.

D. Incorrect - 100°F/Hr is the maximum cooldown rate and not related to the minimum AFW flowrate.

Technical References: DCPP Units 1 & 2 FSAR Update, Revision 19
LSL-5, Plant Cooldown from Minimum Load to Cold Shutdown OP L-5, Revision 11
STG D-1, Auxiliary Feed Water System, Revision 17
LD-1, Auxiliary Feed Water System, Revision 14

References to be provided to applicants during exam: None

Learning Objective: 8430 - Explain significant Auxiliary Feed Water System design features and the importance to nuclear safety.

Question Source: Bank #

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of bus power supplies to the following: Major system loads	Tier #	2
	Group #	1
	K/A #	062 K2.01
	Rating	3.3

Question 43

A loss of the Unit 1 12 kV Bus D would result in a loss of power to which of the following components?

1. RCP 1-1
2. RCP 1-2
3. Circulating Water Pump 1-1
4. Circulating Water Pump 1-2

A. 1 and 3

B. 1 and 4

C. 2 and 3

D. 2 and 4

Proposed Answer: C

Explanation:

A. Incorrect. RCP 1-1 is powered from bus E.

B. Incorrect. RCP 1-1 is powered from bus E.

C. Correct.

D. Incorrect. Circulating Water Pump 1-2 is powered from bus E.

Technical References: Systems Lessons Reactor Coolant Pump and Circulating Water

References to be provided to applicants during exam: None

Learning Objective: State the power supplies to Circulating Water System components. (8346)
State the power supplies to RCP components. (6080)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: EDG	**Tier #**	2
	Group #	1
	K/A #	K3.01
	Rating	3.7*

Question 44

Unit 2 is at 100% power and EDG 2-2 has been running in parallel with off-site power for the last two hours.

Then, DC Control power is lost to EDG 2-2.

The loss of DC Control Power will result in EDG 2-2:

A. Air Start solenoid valves closing.

B. RPM indication being lost.

C. Losing excitation voltage to the generator.

D. Frequency rising to 63.3 Hz.

Proposed Answer: B

Explanation:

A. Incorrect – Air start solenoids close during the starting sequence. Therefore the valves will already be closed at the time of the loss of DC.

B. Correct – Loss of DC control power will result in the loss of EDG RPM and running indications until switched to the alternate power source. Loss of the 125VDC power supply will result in the loss of the EDGs Tachometer Pack, Voltage Regulator (Manual), Alarms, Indications and the Shutdown Relay.

C. Incorrect – Excitation voltage is provided by self excitation from the EDG output while running and loss of DC Control Power will result in the loss of the Field Flash used at startup.

D. Incorrect – Frequency can not increase when paralleled to off-site power.

Technical References: J6B, Diesel Generator System, Revision 21
J9, DC Power, Revision 17

References to be provided to applicants during exam: None

Learning Objective: 37724 - Describe controls, indications, and alarms associated with the Diesel Generator System.

Question Source: Bank #

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Process Rad Monitors. Ability to manually operate and/or monitor in the control room: Effluent release	**Tier #**	2
	Group #	1
	K/A #	073 A4.01
	Rating	2.7

Question 45

While conducting a planned effluent release from Gas Decay Tank 1-1, Unit 1 received alarm PK 11-21, High Radiation.

As a result of this valid alarm;

1. Plant Vent Valve, RCV-17, will close.
2. Unit1/Unit 2 Crosstie valve, FCV-417, will close.
3. Gas Decay Tank Outlet Header to Plant Vent Valve, FCV-410 will close.
4. RE-22, Gas Decay Tank to Plant Vent Radiation Monitor, must be reset to open Plant Vent Valve, RCV-17.

A. 1 and 2

B. 3 and 4

C. 2 and 3

D. 1 and 4

Proposed Answer: D

Explanation:

A. Incorrect. PK 11-21, High Radiation was received due to RM-22, Gas Decay Tank to Plant Vent Radiation Monitor reaching the high alarm set point. Plant Vent Valve, RCV-17, automatically shuts on a high radiation alarm from RM-22. However, FCV-417, Unit 1/Unit 2 Crosstie valve is open. FCV-417 shuts when Gas Decay Tank 1-3 or 2-3 is selected for venting or purging.

B. Incorrect. Gas Decay Tank Outlet Header to Plant Vent Valve, FCV-410, is key operated and administratively controlled. It would be open during an effluent release.

C. Incorrect. FCV-417 shuts when Gas Decay Tank 1-3 or 2-3 is selected for venting or purging. FCV-410, is key operated and administratively controlled and would be open during an effluent release.

D. Correct. Plant Vent Valve, RCV-17, automatically shuts on a high radiation alarm from RM-22. RE-22 must be reset for RCV-17 to open.

Technical References: Systems Lesson Gaseous Radwaste; Operations Lesson Gaseous Radwaste; AR PK 11-21 High Radiation

References to be provided to applicants during exam:		None
Learning Objective: Analyze automatic features and interlocks associated with the Gaseous Radwaste System. (37707)		
Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.7 / 45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect of a loss or malfunction of the following will have on the EDG system: Air Receivers	**Tier #**	2
	Group #	1
	K/A #	064 K6.07
	Rating	2.7

Question 46

An operator reports that EDG 1-1 has an un-isolable air leak resulting in the depressurization of the 1-1A Air Receiver and that 1-1B Air Receiver pressure is 230 psig.

Which of the following will result from this condition?

A. The Starting Air system for EDG 1-1 does not meet the required air capacity for three successive DG start attempts

B. Fuel Oil Day Tank Fill Valve 1-LCV-85, Header B LCV fails closed ONLY

C. Fuel Oil Day Tank Fill Valve 1-LCV-88, Header A LCV fails closed ONLY

D. Fuel Oil Day Tank Fill Valves 1-LCV-88, Header A LCV, and 1-LCV-85, Header B LCV BOTH fail closed

Proposed Answer: B

Explanation:

A. Incorrect – Only one air receiver with pressure greater than 180 psig maintains sufficient air volume for three successive start attempts per SR 3.8.3.4.

B. Correct – Air Receiver 1-1A supplies control air to 1-LCV-85 Header B LCV.

C. Incorrect - Receiver 1-1B supplies control air to 1-LCV-88 Header A LCV.

D. Incorrect - Receiver 1-1B supplies control air to 1-LCV-88 Header A LCV.

Technical References: J6B, Diesel Generator, Revision 213
TS 3.8.3/B 3.8.3, Diesel Fuel Oil, Lube Oil, Starting Air and Turbocharger Air Assist, Amendment No. 135

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content: 55.41.7

Examination Outline Cross-Reference	Level	RO
Service Water. Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Automatic opening features associated with SWS isolation valves to CCW heat exchanges	**Tier #**	2
	Group #	1
	K/A #	076 K4.03
	Rating	2.9

Question 47

GIVEN:

- Unit 1 is at 100% power
- Auxiliary Saltwater Pump (ASW) 1-2 is running
- ASW Pump 1-1 is in standby
- ASW Pump Discharge Cross-Connect Valves, FCV-495 and FCV-496 are open
- Unit 1 and Unit 2 Cross-Tie Valve, FCV-601, is closed
- FCV-602, ASW/CCW Heat Exchanger #1 Inlet Valve was cleared, jacked closed and the air supply isolated

An operator when removing the clearance on FCV-602, unjacks the valve but does not re-establish the air supply. Shortly after, alarm PK 01-01 ASW SYS HX DELTA P/HDR PRESS alarms

With these conditions, FCV-602;

A. is shut and ONLY ASW pump 1-2 is running.

B. is open and BOTH ASW pumps are running.

C. must be in the same position as FCV-603 with only 1 pump running.

D. will shut after a 10 second time delay and the operating pump will switch to Startup Transformer power

Proposed Answer: B

Explanation:

A. Incorrect. The ASW/CCW HX inlet valves are air operated and fail open. Without Instrument air or the backup air supply, the valves will fail open. It is true that the operating ASW pump remains running but the standby pump will also start on low pressure at 40.5 psig.

B. **Correct.** FCV-602 will be open; without Instrument air or the backup air supply, the valves will fail open. The standby pump will have started on low system pressure.

C. Incorrect. During normal operations, only one of the ASW/CCW HX Inlet valves will be open. Both ASW pumps will be running.

D. Incorrect. There is no time delay associated with the operation of the ASW/CCW HX Inlet valves. The operating pump will auto transfer to Startup power after a Safety

Injection signal.

Technical References: Operations Lesson Auxiliary Saltwater System; Systems Lesson Auxiliary Saltwater System; DCPP-AR A0662367

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the ASW System. (5365)

Question Source: (note changes; attach parent)	Bank # Modified Bank # **New**	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	CFR: 41.7	

Examination Outline Cross-Reference	Level	RO
Ability to explain and apply system limits and precautions associated with Instrument Air	**Tier #**	2
	Group #	1
	K/A #	078 2.1.32
	Rating	3.8

Question 48

Air compressors _______________ are controlled from the Unit 2 Control Room and they will automatically start at _________ instrument air pressure if the Compressor Control switch is in AUTO.

A. 0-1 and 0-2; 98 psig and decreasing

B. 0-3 and 0-4; 98 psig and decreasing

C. 0-1 and 0-2; 93 psig and decreasing

D. 0-3 and 0-4; 93 psig and decreasing

Proposed Answer: D

Explanation:

A. Incorrect – Compressors 0-1 and 0-2 are controlled from Unit 1 and 98 psig is the pressure at which the compressor will load if the compressor Control switch is in the ON position and the Master Unloader ON.

B. Incorrect - Compressors 0-3 and 0-4 are controlled from Unit 2 but 98 psig is the pressure at which the compressor will load if the compressor Control switch is in the ON position and the Master Unloader ON.

C. Incorrect - Compressors 0-1 and 0-2 are controlled from Unit 1 but the compressor will start if the Compressor Control switch is in AUTO at 93 psig.

D. Correct - Compressors 0-3 and 0-4 are controlled from Unit 2 and the compressor will start if the Compressor Control switch is in AUTO at 93 psig

Technical References: STG K-1, Compressed Air System, Revision 15

References to be provided to applicants during exam: None

Learning Objective:

Question Source: Bank #
(note changes; attach parent) Modified Bank #
New X

Question History: Last NRC Exam NA

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Containment: **Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Containment evacuation (including recognition of the alarm)**	**Tier #**	2
	Group #	1
	K/A #	103 A2 04
	Rating	3.5

Question 49

While offloading irradiated fuel, the Refueling SRO observes a fuel bundle being removed from the core that is severely damaged.

Which of the following identifies required actions regarding containment as listed in OP AP-21, Irradiated Fuel Damage?

A. Fuel Handling Building (FHB) Ventilation is transferred to Iodine Removal Mode of Operation; check Containment Purge Exhause (RCV-11 and RCV-12) are closed.

B. Start Containment Vent Fans S-3 and E-3; make the Equipment Hatch capable of being closed and held in place by four bolts.

C. Activate CONTMT evacuation alarm; check Containment Purge Supply valves (FCV-660 and FCV-661) are closed; and ensure at least one door in the air lock is shut.

D. PA announcement to "establish Containment Closure;" evacuate unnecessary refueling personnel from Containment.

Proposed Answer: C REPLACE WITH DC QUESTION PROVIDED

Explanation:

A. Incorrect. FHB ventilation mode change is for accidents in the FHB. RCV-11 and RCV-12 are part of checking containment isolations closed.

B. Incorrect. OP AP-21 directs operators to Initiate Containment Closure per Containment Closure Procedure AD8DC.54, which gives instruction to Secure fans S-3 and E-3 to prevent Containment Differential Pressure from interfering with closure of the equipment hatch. Also, in order to move fuel the Equipment Hatch must already be capable of being closed and held in place by four bolts.

C. Correct. OP AP-21 directs the evacuation alarm activated and FCV-660 and FCV-661 checked closed as part of verifying Containment Isolation.

D. Incorrect. In accordance with AD8DC.54 as directed by OP AP-21 this is the correct PA after activating the Containment Evacuation alarm for 10 seconds. OP AP-21 states Evacuate personnel from Containment and does not include the word "unnecessary." While AD8DC.54 does state that "all personnel not associated with the containment closure team," evacuate - all refueling personnel must evacuate and core alterations stop.

Technical References: Technical Specification 3.9.4 Refueling Operations; Systems Lesson

Fuel Handling; Operations Lesson Fuel Handling.

References to be provided to applicants during exam:		None
Learning Objective: Describe system interrelationships between the Fuel Handling System and other plant systems. (40544)		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 41.5 / 43.5 / 45.3 / 45.13	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Closed cooling water flow rate and temperature	**Tier #**	2
	Group #	1
	K/A #	005 A1.03
	Rating	2.5

Question 50

GIVEN:

- Unit 2 is in Mode 6
- RHR pump 1-1 and both RHR Heat Exchangers are in service for decay heat removal.
- CCW pumps 1-1 and 1-2 are operating.
- FI-970 A, B RHR Cold Leg 1 & 2 flow and FI-971, A, B RHR Cold Leg 3 & 4 flow indicate 5000 GPM
- TE-639, HX 1 Outlet Temperature and TE-649, HX 2 Outlet Temperature indicate 287°F

If one CCW pump is lost, the operator would throttle _________ HCV-670, RHR heat Exchanger Bypass valve, and throttle __________ HCV-637/638, RHR HX Outlet valve in order to maintain the same cooldown rate while maintaining the flowrate below the limit in OP B-2:V, RHR Place in Service.

A. open: open

B. open; shut

C. shut; open

D. shut; shut

Proposed Answer: C

Explanation:

A. Incorrect – Throttling open HCV-670 will bypass the heat exchanger and limit cooldown rate in addition opening HCV-670 will exceed the 5000 GPM maximum flowrate to the cold legs

B. Incorrect - Throttling open HCV-670 will bypass the heat exchanger and shutting HCV-637/638 will reduce the flow through the heat exchanger resulting in a smaller cooldown rate.

C. Correct – Throttling shut HCV-670 while opening HCV-637/638 will maintain the flowrate to the loops below the 5000 GPM limit in addition to increasing flow through the RHR HX to maintain the same RCS cooldown rate due to the reduced CCW flow to the RHR HX.

D. Incorrect – Throttling HCV-670 and HCV 637/638 shut will limit the flow through the RHR heat exchanger and reduce total flow to the RCS cold legs resulting in a reduced cooldown rate.

Technical References: OP B-2:V, RHR – Place in Service, Revision 34
STG B2, Residual Heat Removal System, Revision 19
OP L-5, Plant Cooldown from Minimum Load to Cold Shutdown, Revision 89

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS (Main and Reheat Steam)	**Tier #**	2
	Group #	1
	K/A #	039 A3.02
	Rating	3.1

Question 51

GIVEN:

- Unit 1 is at 100% power
- Power has been lost to 120 VAC bus PY-12.
- PC 514C and PC 515C indicate steam generator pressure is 550 psig.

Given these conditions;

1. Reheat Stop and Intercept Valves are closed
2. STM LEAD ISOL 3 and STM LEAD ISOL 4, FCV-44 and FCV-43, are open
3. SG 1-1 and SG 1-2 BD ISOL valves, FCV-760 and FCV-761, are closed
4. SG 1-1 and SG 1-2 ISOL OUTSIDE CONTMT, FCV-151 and FCV-154, are closed

A. 1 and 2

B. 2 and 3

C. 1 and 3

D. 2 and 4

Proposed Answer: C

Explanation:

A. Incorrect. (1) Correct: Reheat Stop and Intercept valves are closed on the turbine trip, which resulted from the Low Steam Line Pressure Safety Injection Signal and from the turbine trip due to loss of load. (2) Incorrect: Main Steam Line Isolation Valves are all closed on a Main Steam Line Isolation, which resulted from 2/3 channels on 1 SG ≤600 psig.

B. Incorrect. (2) FCV-43/44 are closed. (3) Correct: SG blowdown isolation valves inside of containment are closed on a Main Steam Line Isolation.

C. Correct. (1) Correct: Reheat Stop and Intercept valves are closed on the turbine trip, which resulted from the Low Steam Line Pressure Safety Injection Signal and from the turbine trip due to loss of load. (3) Correct: SG blowdown isolation valves inside of containment are closed on a Main Steam Line Isolation.

D. Incorrect. (2) Incorrect: Main Steam Line Isolation Valves are all closed on a Main Steam Line Isolation, which resulted from 2/3 channels on 1 SG ≤600 psig. (4) Only SG blowdown isolation valves INSIDE of containment are closed on a Main Steam Line Isolation.

Technical References: Operations Lesson Reactor Protection System; System Lesson Main Steam System; Operations Lesson Main Steam System

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Main Steam System. (7340)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.5 / 45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of MFW design feature(s) and/or interlock(s) which provide for automatic feedwater isolation of MFW	Tier #	2
	Group #	1
	K/A #	059 K4.19
	Rating	3.2

Question 52

GIVEN:

- Unit 1 tripped from full power
- RCS Loop average temperature:
  - Loop 11: 552°F and lowering slowly
  - Loop 12: 557°F and lowering slowly
  - Loop 13: 554°F and lowering slowly
  - Loop 14: 555°F and lowering slowly
- Pressurizer Pressure is 1850 psig on all channels and stable
- Steam Generator Narrow Range Level:
  - Loop 11: 72% and rising slowly
  - Loop 12: 75% and rising slowly
  - Loop 13: 74% and rising slowly
  - Loop 14: 73% and rising slowly
- A Main Feedwater Isolation has occurred

Which of the following signals could have caused the Main Feedwater Isolation?

A. SI ONLY

B. SI or P-4 coincident with Low RCS Tave

C. SI or P-14

D. P-4 coincident with Low RCS Tave ONLY

Proposed Answer: A

Explanation:

A. Incorrect – The SI actuated at primary pressure of 1850 and initiated feedwater isolation. But answer does not include the correct answer of the P-4 interlock.

B. Incorrect – The P-4 interlock actuates following a Rx trip with 2/4 average coolant temperatures ≤ 554°F. The SI actuated at primary pressure of 1850 and initiated feedwater isolation.

C. Incorrect - The P-14 interlock requires 1/4 Steam Generators level to be > 75% (U2 >90%) narrow range. The P-4 interlock actuates following a Rx trip with 2/4 average coolant temperatures ≤ 554°F.

D. Incorrect – The P-4 interlock actuates following a Rx trip with 2/4 average coolant temperatures ≤ 554°F. But answer does not include the correct answer of the SI actuation.

Technical References: AR PK09-11, AR PK09-12, OIM B-6-12, LC-8A, LC-6A

References to be provided to applicants during exam: None

Learning Objective: 37614 - Analyze automatic features and interlocks associated with the Main Feedwater System.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the AFW will have on the following: S/G	**Tier #**	2
	Group #	1
	K/A #	061 K3.02
	Rating	4.2

Question 53

Following a Loss of Offsite Power, Unit 1 tripped from 100% power and Safety Injection actuated. While performing EOP E-0.1, Reactor Trip Response, Auxiliary Feedwater flow indicates 0 gpm and all four Steam Generator Narrow Range Levels are approximately 9% and lowering.

With these conditions;

A. The Steam Generators (SGs) will remove decay heat less efficiently but will retain their function as the heat sink. Cool down the Reactor Coolant System and place the Residual Heat Removal System on service.

B. A Small Break Loss of Coolant Accident through the RCP seals is the highest priority safety concern. Restore off site power

C. A Steam Generator low level condition following a reactor trip is an expected condition. Auxiliary Feedwater flow will restore SGWL to within the NR without further operator action.

D. The SGs will lose the ability to remove decay heat as their inventory depletes; restore Auxiliary Feedwater Flow or establish RCS Bleed and Feed.

Proposed Answer: D

Explanation:

A. Incorrect. Without secondary side inventory, the SGS will lose the ability to function as a heat sink. Placing RHR on service would take too long, restore AFW or Bleed and Feed.
B. Incorrect. The highest priority safety concern is restoring a heat sink to remove decay heat. Restoring Offsite Power will not restore the heat sink.
C. Incorrect. SG levels are expected to be low following a trip, but per E-0 Reactor Trip, if feed water flow is less than 435 gpm and level is , 15%, then operators should go to the functional recovery for a Loss of Heat Sink.
D. **Correct.** If reactor trip and SI occur, the operation of the AFW system is verified in guideline E-0, REACTOR TRIP OR SAFETY INJECTION, prior to Status Tree monitoring. If minimum AFW flow is not being provided, the operator is directed to implement guideline FR-H.1.

Technical References: E-0, Reactor Trip or Safety Injection; FR-H.1, Response to Loss of Heat Sink Background Document

References to be provided to applicants during exam: None

Learning Objective:	State the minimum AFW flow required to maintain S/Gs as a heat sink. (6982)	
Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	CFR: 41.7 / 45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect of a loss or malfunction of the fuel oil storage tanks will have on the EDG system	**Tier #**	2
	Group #	1
	K/A #	064 K6.08
	Rating	3.2

Question 54

EDG 11 is running fully loaded and the fuel oil day tank is full when Fuel Oil Transfer Pump 0-2 trips.

If the day tank is not refilled, the EDG would be expected to run for approximately ________ hour(s) before running out of fuel.

A. One-half.

B. one.

C. Two and one-half.

D. Twenty-four.

Proposed Answer: C

Explanation:

A. Incorrect – The day tank hold enough fuel for approximately two and one-half hours at full load.

B. Incorrect - The day tank hold enough fuel for approximately two and one-half hours at full load.

C. Correct - The day tank hold enough fuel for approximately two and one-half hours at full load.

D. Incorrect - The day tank hold enough fuel for approximately two and one-half hours at full load.

Technical References: LJ-6B, Diesel Generator System, Revision 15

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X

Comprehensive/Analysis

10CFR Part 55 Content: 55.41.7

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause effect relationships between the containment system and the following systems: CCS	**Tier #**	2
	Group #	1
	K/A #	103 K1.01
	Rating	3.6

Question 55

Initial Conditions:

- Unit 1 is at 100% power
- Containment Fan Cooling Units (CFCU) 1-1, 1-2, and 1-3 are in service
- Component Cooling Water Heat Exchanger outlet temperatures are normal
- Containment Environment PPC alarm, PK 01-16, comes in
- containment temperature is 110°F and slowly rising

In this situation, the operators should:

A. Place an additional CFCU on service.

B. Initiate a plant shutdown.

C. Increase CCW flow to maximum available.

D. Secure one or more CRDM fan coolers and lower reactor power level.

Proposed Answer: A

Explanation:

A. Correct. Place an additional CFCU on service, this will help cool Containment.

B. Incorrect. The technical Specification limit of 120°F has not been reached; at which point, operators have 8 hours to restore temperature to within limits.

C. Incorrect. The CCW HX outlet temperatures are normal, therefore increasing CCW flow to the CFCUs is not necessary. The problem is that there are not enough CFCUs on service; not the temperature of the cooling water.

D. Incorrect. If one or more CRDM fan coolers were secured, then the correct action would be to start one or more, not to secure them.

Technical References: Containment Environment PPC alarm, PK 01-16; Operations Lesson Containment Fan Cooling Units, LH-2.

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the CFCUs. (40812)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	CFR: 41.2 to 41.9 / 45.7 to 45.8	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of a loss of pressurizer level (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of the loss of pressurizer level	**Tier #**	2
	Group #	2
	K/A #	011 A2.03
	Rating	3.8

Question 56

Unit 1 is at 100% power with CCP 1-1 operating and normal letdown is in-service.

The operator observes pressurizer level is 35% and lowering and charging flow rate is 150 GPM. The SFM directs you to take actions in accordance with AP-1, Excessive Reactor Coolant System Leakage.

The sequence of actions to isolate the leak would be to (1) and if pressurizer level reaches the PK05-22, Pressurizer Level Hi/Lo Control, low level set point then (2).

A. (1) Start an additional charging pump then isolate letdown; (2) letdown automatically isolates.

B. (1) Start an additional charging pump then isolate letdown; (2) all pressurizer heaters de-energize.

C. (1) Isolate letdown then start an additional charging pump; (2) automatic isolation of letdown.

D. (1) Isolate letdown then start an additional charging pump; (2) all pressurizer heaters de-energize.

Proposed Answer: A

Explanation:

A. Correct – Order of actions is correct in accordance with OP AP-1. When pressurizer level reaches the PK05-22 low level setpoint (17%) letdown is automatically isolated in addition to de-energizing all pressurizer heaters with the exception of vital powered heaters in accordance with PK05-22.

B. Incorrect - Order of actions is correct in accordance with OP AP-1. De-energizing all pressurizer heaters is not correct. Vital powered pressurizer heaters are not de-energized at 17% pressurizer level.

C. Incorrect – Order of actions are reversed in accordance with OP AP-1. When pressurizer level reaches the PK05-22 low level setpoint (17%) letdown is automatically isolated in addition to de-energizing all pressurizer heaters with the exception of vital powered heaters in accordance with PK05-22.

D. Incorrect - Order of actions are reversed in accordance with OP AP-1. De-energizing all pressurizer heaters is not correct. Vital powered pressurizer heaters are not de-energized at 17% pressurizer level.

Technical References: OP AP-1, Excessive Reactor Coolant System Leakage, Revision 20

PK05-22, Pressurizer Level Hi/Lo Control, Revision 6

References to be provided to applicants during exam:		None
Learning Objective:		
Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause effect relationships between the NIS and the following systems: RPS	Tier #	2
	Group #	2
	K/A #	015 K1.01
	Rating	4.6

Question 57

GIVEN:

- Unit 1 is performing a reactor startup
- Rx power is at $1x10^{-11}$ amps
- Instrument Bus PY-11 loses power
- The NI-31 Level Trip Switch is in Normal.

As a result;

A. The reactor will not trip since gamma compensation is still enabled.

B. The reactor will trip.

C. Only Train A of the Reactor Protection System calls for a trip.

D. No auto actions will occur; however, Technical Specification actions will be required since the channel is required to be operable.

Proposed Answer: B

Explanation:

A. Incorrect. The reactor will trip regardless of whether or not gamma compensation is enabled.

B. Correct. On a loss of instrument power with the Level Trip Bypass Switch in Normal, the reactor will trip.

C. Incorrect. Both trains of RPS will call for a reactor trip.

D. Incorrect. The reactor will trip.

Technical References: OIM drawing B-4-2 Excore Nuclear Instrumentation; Operations Lesson LB-4, Excore Nuclear Instrumentation

References to be provided to applicants during exam: None

Learning Objective: Describe system interrelationships between the Excore Nuclear Instrumentation System and other plant systems. (36973)

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	

10CFR Part 55 Content: CFR: 41.2 to 41.9 / 45.7 to 45.8

Examination Outline Cross-Reference	Level	RO
Ability to manually operate and/or monitor the Non-nuclear instrumentation channel select controls in the control room	**Tier #**	2
	Group #	2
	K/A #	016 A4.01
	Rating	2.9

Question 58

Unit 1 is at full power.

RCS pressure channels indicate as follows:

- PT – 455 – 2242 psig
- PT – 456 – 2235 psig
- PT – 457 – 2238 psig
- PT – 474 – 2244 psig

Which pressure signal will be used by the PCS for pressurizer pressure control?

A. PT-455

B. PT-456

C. PT-457

D. PT-474

Proposed Answer: A

Explanation:

A. Correct – Pressurizer pressure control is maintained by the second highest pressure signal.

B. Incorrect - Pressurizer pressure control is maintained by the second highest pressure signal.

C. Incorrect - Pressurizer pressure control is maintained by the second highest pressure signal.

D. Correct - Pressurizer pressure control is maintained by the second highest pressure signal.

Technical References: Operator Information Manual A-4-4b, Pressurizer Control Channel Failures, Revision 28
OP A-4A:I, Pressurizer – Make Available, Revision 28

References to be provided to applicants during exam: None

Learning Objective:

Question Source: Bank #

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Ability to interpret and execute procedure steps: In-core Temperature Monitor.	**Tier #**	2
	Group #	2
	K/A #	017 2.1.20
	Rating	4.6

Question 59

The Unit 2 crew is performing the actions in FR-C.1, Response to Inadequate Core Cooling, and are about the start an ECCS charging pump.

Following the start of the ECCS charging pump, INITIALLY the Core Exit Thermocouples (CET) indications will __________.

A. Decrease due to saturated steam forming a frothy two phase mixture.

B. Decrease due to superheated steam forming a frothy two phase mixture.

C. Increase due to saturated steam being forced out of the core.

D. Increase due to superheated steam being forced out of the core.

Proposed Answer: D

Explanation:

A. Incorrect. Inititally CET indication will Increase, not decrease. As the core begins to fill, heat transfer from the fuel will cause fluid entering the core to form a frothy two phase mixture.

B. Incorrect. Same as A.

C. Incorrect. Steam exiting the core will be superheated.

D. **Correct.** Increase due to superheated steam exiting the core.

Technical References: F0.2; FR.C-1, Functional Recovery Core Cooling Background pg LPE-C, page 10

References to be provided to applicants during exam: None

Learning Objective: Describe the emergency operating procedure strategies for: (41700) SI Termination and SI Reinitiation

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.10 / 43.5 / 45.12	

Examination Outline Cross-Reference	Level	RO
Knowledge of bus power supplies to the containment iodine removal fans	**Tier #**	2
	Group #	2
	K/A #	027 K2.01
	Rating	3.1

Question 60

Which of the following states the Bus and MCC that supplies power to Iodine Removal Fan 1-E15?

A. Non-vital 4kV Bus D through MCC 12I

B. Non-vital 4kV Bus D through MCC 12J

C. Non-vital 4kV Bus E through MCC 12I

D. Non-vital 4kV Bus E through MCC 12J

Proposed Answer: C

Explanation:

A. Incorrect – MCC 12I is powered from Bus E.

B. Incorrect - Non-vital 4kV Bus D through MCC 12J powers Iodine Removal Fan 1-E16.

C. Correct - Non-vital 4kV Bus E through MCC 12I powers Iodine Removal Fan 1-E15.

D. Incorrect – MCC 12J is powered from Bus D.

Technical References: OIM J-1-1, Electrical Distribution Overview, Revision 28
OIM J-1-4, Non-vital Electrical Power Distribution Overview, Revision 28
H3, Iodine Removal System, Revision 11

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the Containment Purge System controls including: Radiation levels	**Tier #**	2
	Group #	2
	K/A #	029 A1.02
	Rating	3.4

Question 61

Unit 1 is at 100% power.

The following sequence of events occur:

- AR PK02-06, CONTMT VENT ISOLATION, alarms
- It is determined that the alarm was due to Containment Purge Radiation Monitor, RE-44A, failing high
- The operator resets the CVI signal with RE-44A still in alarm
- All Containment vent isolation components are returned to their normal position

Moments later AR PK02-06, CONTMT VENT ISOLATION, alarms again due to an actual high radiation condition detected by RE-44B.

Based on this sequence of events, currently, Containment Purge System;

1. Fan S-3 is Off
2. Fan E-3 is On
3. Supply Valves, FCV-660 and 661, are shut
4. Exhaust Valves, RCV-11 and 12, are open

A. 1 and 4

B. 2 and 4

C. 2 and 3

D. 1 and 2

Proposed Answer: B

Explanation:

A. Incorrect. Resetting the Containment Ventilation Isolation signal without first clearing the condition(s) that brought in the alarm will inhibit automatic containment ventilation isolation from another high radiation signal. This would require manually actuating CVI. Therefore, Fan S-3 would still be ON.

B. Correct. Without manually activating CVI, Fan E-3 will still be On and the Containment Purge Exhaust Valves will still be open.

C. Incorrect. Containment Purge Supply valves will still be Open.

D. Incorrect. Fan S-3 will still be ON.

Technical References: AR PK02-06, Contmt Vent Isolation; Operations Lesson Containment Purge System.

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Containment Purge System. (5119)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	CFR: 41.5 / 45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of design features which provide spent fuel pool cooling anti-siphon devices	**Tier #**	2
	Group #	2
	K/A #	033 K4.03
	Rating	3.1

Question 62

The reason for having an anti-siphon device on the spent fuel pool is that it ensures greater than ______ of water exists above the top of the fuel assemblies in the event an inadvertent drainage occurs. It is accomplished by the use of a half inch hole in the ________.

A. 10 feet; SFP cooling return line.

B. 23 feet; SFP cooling return line.

C. 10 feet; SFP cooling supply line.

D. 23 feet; SFP cooling supply line.

Proposed Answer: A

Explanation:

A. Correct – UFSAR Section 9.1.3.1.2, states that the design of the spent fuel pool dewatering protection ensures greater than ten feet of water exists over the top of the fuel assemblies should inadvertent draining occur by the use of a siphon breaker on the return line into the spent fuel pool. The siphon break consists of a half inch hole in the cooling return line.

B. Incorrect - UFSAR Section 9.1.3.1.2, states that the design of the spent fuel pool dewatering protection ensures greater than ten feet of water exists over the top of the fuel assemblies should inadvertent draining occur by the use of a siphon breaker on the return line into the spent fuel pool. The normal spent fuel pool water level is 23 feet of water. The siphon break consists of a half inch hole in the cooling return line.

C. Incorrect - UFSAR Section 9.1.3.1.2, states that the design of the spent fuel pool dewatering protection ensures greater than ten feet of water exists over the top of the fuel assemblies should inadvertent draining occur by the use of a siphon breaker on the return line into the spent fuel pool. The siphon break consists of a half inch hole in the cooling return line vs. the supply line.

D. Incorrect - UFSAR Section 9.1.3.1.2, states that the design of the spent fuel pool dewatering protection ensures greater than ten feet of water exists over the top of the fuel assemblies should inadvertent draining occur by the use of a siphon breaker on the return line into the spent fuel pool. The normal spent fuel pool water level is 23 feet of water. The siphon break consists of a half inch hole in the cooling return line vs. the supply line.

Technical References: DCPP UFSAR Revision 19
OIM 106713, Spent Fuel Cooling System, Sheet 3

References to be provided to applicants during exam:		None
Learning Objective:		
Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Fuel Handling Equipment. - Knowledge of physical connections and/or cause-effect relationships between Fuel handling system and the following system: RHR	Tier #	2
	Group #	2
	K/A #	034 K1.02
	Rating	2.5

Question 63

Unit 1 is in Mode 6; there is fuel in the reactor. The Shift Foreman directs you to refill the refueling cavity in accordance with OP B-2, “RHR Filling the Refueling Cavity”, and to maintain the lowest possible dose rate around the Refueling Cavity.

Which of the following is the correct pump and flow path?

A. CS pump 1-1 via the RHR injection line to loops 1 and 2 hot legs.

B. CS pump 1-1 via the RHR injection line to loops 1 and 2 cold legs.

C. RHR pump 1-1 to loops 1 and 2 cold legs.

D. RHR pump 1-1 to loops 1 and 2 hot legs.

Proposed Answer: A

Explanation:

A. Correct. In accordance OP B-2, if fuel is in the reactor, an RHR pump will NOT be used. Directing Cavity flow through the Hot Legs bypasses the lower reactor vessel and the core, and should reduce the amount of activity pumped into the Refueling Cavity, lowering dose rate in the area.

B. Incorrect. Injecting into the cold legs would not minimize activity in the Reactor Cavity and dose rates in the area.

C. Incorrect. If fuel is in the reactor, an RHR pump will NOT be used. Also, injecting into the cold legs would not minimize activity in the Reactor Cavity and dose rates in the area.

D. Incorrect. If fuel is in the reactor, an RHR pump will NOT be used.

Technical References: OP B-2 RHR Filling the Refueling Cavity

References to be provided to applicants during exam: None

Learning Objective: Discuss significant precautions and limitations associated with the Fuel Handling System. (36965)

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.2 to 41.9 / 45.7 to 45.8	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the Steam Dump/Turbine Bypass Control will have on the Reactor Coolant System	**Tier #**	2
	Group #	2
	K/A #	041 K3.02
	Rating	3.8

Question 64

GIVEN:

- Unit 2 has tripped
- EOP E-0, Reactor Trip or Safety Injection, actions are in progress.
- Steam Dump Mode Select Switch (43/SDI) is selected to Tavg
- The Reactor Trip Controller failed following the reactor trip

With no operator action, the steam dump valves should stabilize average reactor coolant temperature at approximately _______.

A. 543°F

B. 547°F

C. 550°F

D. 555°F

Proposed Answer: C

Explanation:

A. Incorrect – P-12 will cause the valves to close at 543°F but the Load Reject Controller will maintain reactor coolant temperature modulating at 550°F with no operator action.

B. Incorrect – Reactor coolant temperature tracks to 547°F when the steam dumps are controlled in Tavg mode by the Reactor Trip controller.

C. Correct – Reactor coolant temperature tracks to 550°F when the steam dumps are controlled in Tavg mode by the Load Reject controller in Unit 2.

D. Incorrect - Reactor coolant temperature tracks to 550°F when the steam dumps are controlled in Tavg mode by the Load Reject controller in Unit 2.

Technical References: OIM C-2-6, Steam Dump System Composite, Revision 29
C2B, Steam Dumps, Revision 17
LC-2B, Steam Dump System, Revision 13

References to be provided to applicants during exam: None

Learning Objective: 37812 - Describe Steam Dump System components.

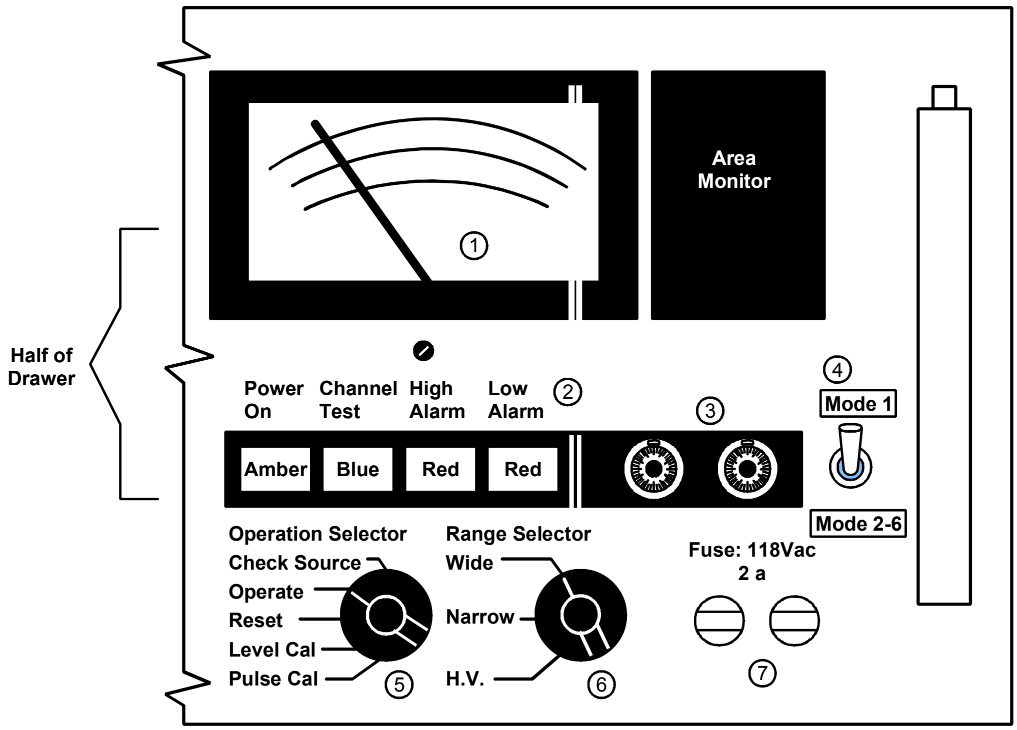
Question Source: Bank #

(note changes; attach parent)	Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Area Radiation Monitoring. Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: Radiation levels	**Tier #**	2
	Group #	2
	K/A #	072 A1.01
	Rating	3.4

Question 65

An operator notices the RM 01 (shown below) Operation Selector is in the “Level Cal” position.



With the Operation Selector in this position, the radiation monitor ________.?

A. Applies a pre-set test voltage for testing alarm set points and indications.

B. Applies an adjustable test voltage for testing alarm set points and indications.

C. Resets the alarm set point to zero in order to allow an operator to manually adjust the set point.

D. Applies a pulse signal to test the alarm circuitry downstream of the detector.

Proposed Answer: B

Explanation:

A. Incorrect. The voltage applied is adjustable, not pre-set.

B. Correct. An adjustable test voltage for testing alarm set points and indications.

C. Incorrect. It does not reset the alarm set point to any value; only allows for testing set points and indications.

D. Incorrect. This is what the Pulse Cal position does.

Technical References: Operations Lesson Radiation Monitoring System

References to be provided to applicants during exam: None

Learning Objective: Describe controls, indications, and alarms associated with the Radiation Monitoring System. (37875)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.5 / 45.5	

Examination Outline Cross-Reference	Level	RO
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	**Tier #**	3
	Group #	-
	K/A #	2.1.7
	Rating	4.4

Question 66

Unit 1 is at 30% power.

An inadvertent dilution will result in automatic rod _________________ due to _________________ RCS temperature.

A. withdrawal; lowering

B. insertion; rising

C. withdrawal; rising

D. insertion, lowering

Proposed Answer: B

Explanation:

A. Incorrect – dilution will result in an RCS temperature increase vice a lowering temperature. The automatic rod withdrawal is consistent with the lowering RCS temperature.

B. Correct - dilution will result in an RCS temperature increase and automatic rod insertion.

C. Incorrect - dilution will result in an RCS temperature increase. The automatic rod withdrawal is consistent with a lowering not rising RCS temperature.

D. Incorrect – dilution will result in an RCS temperature increase vice a lowering temperature. The automatic rod insertion is consistent with a rising not lowering RCS temperature.

Technical References: Instructor Lesson Guide LTAA5, Reactivity Addition Accidents, Revision 10
DCPP U1 and U2 UFSAR, Revision 19

References to be provided to applicants during exam: None

Learning Objective:
Question Source: Bank #
(note changes; attach parent) Modified Bank #
New X

Question History: Last NRC Exam NA

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Ability to direct personnel activities inside the control room.	Tier #	3
	Group #	-
	K/A #	2.1.9
	Rating	2.9

Question 67

The Unit 1 Control Operator sees and unexpected increase in steam flow and reactor power is rapidly approaching the trip setpoint. The Shift Foreman is attending the Shift Brief.

Whose concurrence, if any, is required to trip the plant?

A. The Shift Manager ONLY

B. The Control Operator ONLY

C. Shift Foreman ONLY

D. No concurrence is required

Proposed Answer: D

Explanation:

A. Incorrect. May direct the plant to be tripped, but permission need not come from this position.

B. Incorrect. The CO may also direct the plant to be tripped, but permission does not necessarily have to come from him.

C. Incorrect. The SF may direct the plant to be tripped, but permission need not come from this position.

D. Correct. In accordance with OP1.DC10, Conduct of Operations, Licensed Operators are expected to manually initiate Engineered Safety Feature (ESF) actions, (Reactor Trips and Safety Injections), under the following circumstances: when plant parameters are approaching an automatic set point, such that the automatic action is judged to be unavoidable.

Technical References: OP1.DC10, Conduct of Operations; Operations Lesson Conduct of Operations, LADM-1.

References to be provided to applicants during exam: None

Learning Objective: Describe the general duties and responsibilities of Control Room Operators. (41662)

Question Source: Bank #
(note changes; attach parent) Modified Bank #
New X

Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.10 / 45.5 / 45.12 / 45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of the process for controlling equipment configuration or status.	**Tier #**	3
	Group #	-
	K/A #	2.1.14
	Rating	3.9

Question 68

In accordance with OP1.DC10, Conduct of Operations, Section 5.15, Plant Status Control, the Gaseous Radioactive Waste System is controlled via;

A. pink tags and documentation in the off normal watchstation turnover checklist

B. status board and documentation in the off normal watchstation turnover checklist

C. pink tags and the controlling document

D. status board and the controlling document

Proposed Answer: B

Explanation:

A. Incorrect – The Gaseous Radioactive Waste System configuration control is maintained by the use of a status board and off normal watch station turnover checklist. A pink tag is used for controlling positional components that are not part of the Liquid Radioactive Waste, Gaseous Radioactive Waste or Condensate Polishing Systems.

B. Correct - The Gaseous Radioactive Waste System configuration control is maintained by the use of a status board and off normal watch station turnover checklist.

C. Incorrect - A pink tag and controlling document are used for controlling positional components that are not part of the Liquid Radioactive Waste, Gaseous Radioactive Waste or Condensate Polishing Systems.

D. Incorrect - A pink tag is used for controlling positional components that are not part of the Liquid Radioactive Waste, Gaseous Radioactive Waste or Condensate Polishing Systems. A status board is used to indicate off normal alignments during Special Method Status Control which includes the Gaseous Radioactive Waste System.

Technical References: OP1.DC10, Conduct of Operations, Revision 30

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)

Bank #
Modified Bank #
New X

Question History: Last NRC Exam NA

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of conditions and limitations in the facility license.	Tier #	3
	Group #	-
	K/A #	2.2.38
	Rating	3.6

Question 69

Per the Unit 1 Operating License, Pacific Gas and Electric Company, the maximum allowed reactor thermal power is:

A. 3421 MW_T

B. 3411 MW_T

C. 3401 MW_T

D. 3311 MW_T

Proposed Answer: B

Explanation:

A. Incorrect. 3411 MwT is the maximum allowed power level.

B. Correct. 3411 MwT is the maximum allowed power level.

C. Incorrect. 3411 MwT is the maximum allowed power level.

D. Incorrect. 3411 MwT is the maximum allowed power level.

Technical References: Diablo Canyon Unit 1 Operating License

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 41.7 / 41.10 / 43.1 / 45.13	

Examination Outline Cross-Reference	Level	RO
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	**Tier #**	3
	Group #	-
	K/A #	2.2.44
	Rating	4.2

Question 70

GIVEN:

- Unit 1 is at 45% power
- RCP 1-1 tripped
- PK04-11, Reactor Trip Initiate – ON
- PK04-14, Reactor Trip Actuated – OFF
- PK05-01, RCP No. 11 – ON

In this situation, the operators should:

A. Trip the reactor due to the loss of an RCP below the P-8 set point

B. De-energize 480V Buses 12D and 12E due to failure of the plant to automatically trip

C. De-energize 480V Buses 13D and 13E due to failure of the plant to automatically trip

D. De-energize 480V Buses 14D and 14E due to failure of the plant to automatically trip

Proposed Answer: C

Explanation:

A. Incorrect – Power level is above the P-8 set point of 35%. Therefore, the plant should have automatically tripped.

B. Incorrect – PK04-11 – ON and PK04-14 – OFF are E-0 entry conditions which requires manually de-energizing 480V Buses 13D and 13E instead of 480V Buses 12D and 12E.

C. Correct – PK04-11 – ON and PK04-14 – OFF are E-0 entry conditions which requires manually de-energizing 480V Buses 13D and 13E.

D. Incorrect – PK04-11 – ON and PK04-14 – OFF are E-0 entry conditions which requires manually de-energizing 480V Buses 13D and 13E instead of 480V Buses 14D and 14E.

Technical References: E-0, Reactor Trip or Safety Injection, Revision 40
PK04-11, Reactor Trip Initiate, Revision 16
PK04-14, Reactor Trip Actuated, Revision 8
PK05-01, RCP NO. 11, Revision, 33

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	**Tier #**	3
	Group #	-
	K/A #	2.3.5
	Rating	2.9

Question 71

For portable radiation monitoring instruments, the maximum allowed difference between the instrument response reading and the source is ______.

A. ±10%

B. ±15%

C. ±20%

D. ±25%

Proposed Answer: C

Explanation:

A. Incorrect. In accordance with RCP D-900, Performance Tests for Radiation Protection Instruments, the tolerance is ±20%.

B. Incorrect. In accordance with RCP D-900, Performance Tests for Radiation Protection Instruments, the tolerance is ±20%.

C. Correct. In accordance with RCP D-900, Performance Tests for Radiation Protection Instruments, the tolerance is ±20%.

D. Incorrect. In accordance with RCP D-900, Performance Tests for Radiation Protection Instruments, the tolerance is ±20%.

Technical References: RCP D-900, Performance Tests for Radiation Protection Instruments

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 41.11 / 41.12 / 43.4 / 45.9	

Examination Outline Cross-Reference	Level	RO
Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	**Tier #**	3
	Group #	-
	K/A #	2.3.13
	Rating	3.4

Question 72

The DCPP Emergency Exposure Guidelines state the dose limit for saving plant equipment is _____. TEDE.

A. 5 Rem

B. 10 Rem

C. 25 Rem

D. 75 Rem

Proposed Answer: B

Explanation:

A. Incorrect – In accordance with EP RB-2, DCPP Emergency Exposure Guidelines, 10 Rem TEDE exposure is the limit for Property Saving. Five rem is the normal limit

B. Correct – In accordance with EP RB-2, DCPP Emergency Exposure Guidelines,10 REM TEDE is the whole body limit for exposure.

C. Incorrect – In accordance with EP RB-2, DCPP Emergency Exposure Guidelines, 25 Rem limit for Life Saving to Individual.

D. Incorrect - In accordance with EP RB-2, DCPP Emergency Exposure Guidelines, 75 Rem limit is for Dose Saving to the Population.

Technical References: Operational Phase Lesson Plan LEP-3, EP RB Procedures, Revision 8
EP RB-2, DCPP Emergency Exposure Guidelines, Revision 8

References to be provided to applicants during exam: None

Learning Objective: 7954 - State the emergency dose limits

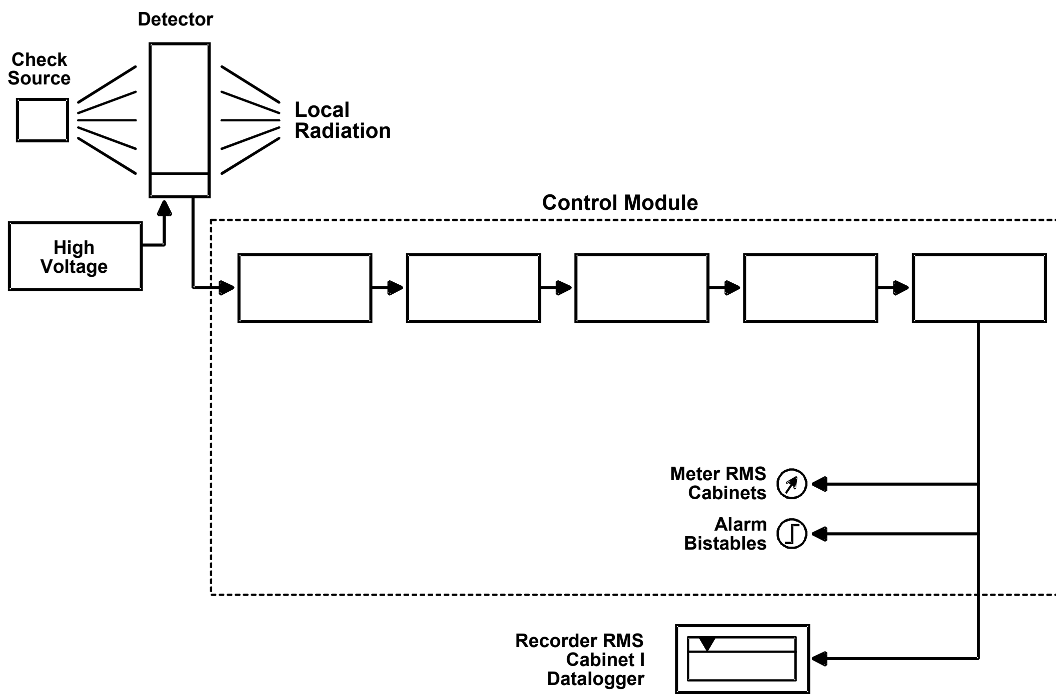
Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	

10CFR Part 55 Content: 55.41.12

Examination Outline Cross-Reference	Level	RO
Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	**Tier #**	3
	Group #	-
	K/A #	2.3.15
	Rating	2.9

Question 73

Shown below is a typical analog radiation monitoring channel.



Which of the following is the correct sequence of the signal path?

A. Pulse Shaper, Diver Amplifier, Discriminator, Log Pulse Integrator, Level Amplifier

B. Pulse Shaper, Discriminator, Level Amplifier , Log Pulse Integrator, Driver Amplifier

C. Discriminator, Pulse Shaper, Driver Amplifier, Log Pulse Integrator, Level Amplifier

D. Driver Amplifier, Discriminator, Pulse Shaper, Log Pulse Integrator, Level Amplifier

Proposed Answer: C

Explanation:

A. Incorrect. The correct sequence is Discriminator, Pulse Shaper, Driver Amplifier, Log Pulse Integrator, Level Amplifier

B. Incorrect. Same.

C. Correct**.** Same.

D. Incorrect. Same.

Technical References: Systems Lesson G4A, Radiation Monitoring Systems

References to be provided to applicants during exam: None

Learning Objective: Identify the location of components associated with the Radiation Monitoring System. (6893)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/**Fundamental** Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.12 / 43.4 / 45.9	

Examination Outline Cross-Reference	Level	RO
Knowledge of operational implications of EOP warnings, cautions, and notes.	**Tier #**	3
	Group #	-
	K/A #	2.4.20
	Rating	3.8

Question 74

The function of a _______________ in an EOP is to provide _____________________________.

A. Caution; information which supports operator action

B. Note; conditional statements that provide for prompt operator response

C. Note; information about potential hazards to personnel or equipment

D. Caution; information about potential hazards to personnel or equipment

Proposed Answer: D

Explanation:

A. Incorrect – Cautions are information about potential hazards to personnel or equipment. The description contained in this answer describes a Note.

B. Incorrect – This statement describes a foldout page, not a Note.

C. Incorrect - Notes are information which supports operator action. The description contained in this answer describes a Caution.

D. Correct Cautions are information about potential hazards to personnel or equipment. The description contained in this answer describes a Note.

Technical References: LPE-Rule, EOP Rules of Usage, Revision 11

References to be provided to applicants during exam: None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of EOP entry conditions and immediate action steps.	**Tier #**	3
	Group #	-
	K/A #	2.4.1
	Rating	4.6

Question 75

Which of the following procedures contain immediate action steps that can be performed by the operator at the controls from memory?

A. - E-0, Reactor Trip/Safety Injection
- ECA-0.0, Loss of All Vital AC
- EOP-0.1, Reactor Trip Response

B. - E-0, Reactor Trip/Safety Injection
- FR-S.1, Response to Nuclear Power Generation/ATWS
- AP-17, Loss of Charging

C. - E-0, Reactor Trip/Safety Injection
- EOP E-1, Loss of Reactor or Secondary Coolant
- AP-12A, Continuous Insertion or Withdrawal of a Control Rod Bank

D. - ECA-0.0, Loss of All Vital AC
- AP-15, Loss of Feedwater
- FR-S.1, Response to Nuclear Power Generation/ATWS

Proposed Answer: D

Explanation:

A. Incorrect. EOP-0.1 does not have immediate action steps.

B. Incorrect. AP-17 does not have immediate action steps.

C. Incorrect. EOP E-1 and AP-12A do not have immediate action steps.

D. **Correct**. All of these have immediate action steps.

Technical References: Operations Lesson LPE Rule – EOP Rules of Usage; E-0, Reactor Trip/Safety Injection; ECA-0.0, Loss of All Vital AC; EOP-0.1, Reactor Trip Response; FR-S.1, Response to Nuclear Power Generation/ATWS; AP-17, Loss of Charging; EOP E-1, Loss of Reactor or Secondary Coolant; AP-12A, Continuous Insertion or Withdrawal of a Control Rod Bank; AP-15, Loss of Feedwater.

References to be provided to applicants during exam: None

Learning Objective: Describe the expectations and standards for abnormal procedure use and adherence, including: Performance of immediate actions (41678)

Question Source: Bank #

(note changes; attach parent)	Modified Bank # New	 X
Question History:	Last NRC Exam	NA
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.10 / 43.5/45.13	

Examination Outline Cross-Reference	Level	SRO
Knowledge of abnormal condition procedures related to small break LOCA	**Tier #**	1
	Group #	1
	K/A #	000009 2.4.11
	Rating	4.2

Question 76

Unit 1 is in Mode 4 during a scheduled outage with the following conditions:

- Pressurizer level is 20% and lowering.
- RCS subcooling is 15°F
- The PZR PORVs and Safeties are all closed
- Train 'A' RHR is aligned for shutdown cooling.

Which of the following procedures should be entered?

E. EOP E-1, Loss of Reactor or Secondary Coolant

F. OP AP-24, Shutdown LOCA

G. OP AP SD-0, Loss of, or Inadequate Decay Heat Removal

H. OP AP SD-2, Loss of RCS Inventory

Proposed Answer: B

Explanation:

A. Incorrect – EOP E-1 used for actions following events while at power.

B. Correct – Applicable to Mode 4 for loss of RCS inventory.

C. Incorrect – Applicable to Mode 5 or 6.

D. Incorrect– Applicable to Mode 5 or 6.

Technical References: EOP E-1, Loss of Reactor or Secondary Coolant, Revision 30
OP AP-24, Shutdown LOCA, Revision 10
OP AP SD-2, Loss of RCS Inventory, Revision 18
OP AP SD-0, Loss of, or Inadequate Decay Heat Removal, Revision 12

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to determine or interpret the following as they apply to a Large Break LOCA: That equipment necessary for functioning of critical pump water seals is operable	Tier #	1
	Group #	1
	K/A #	011 EA2.07
	Rating	3.2

Question 77

A Large Break Loss of Coolant Accident has occurred on Unit 1 and currently:

- Containment Pressure is 22 psig and slowly rising
- EOP F0.2, Core Cooling, is RED
- EOP FR-C.1, Response to Inadequate Core Cooling, is the procedure in effect

In accordance with EOP FR-C.1, the Shift Foreman should:

A. Direct operators to go to Appendix B and restart an RCP before continuing with the remaining steps of EOP FR-C.1

B. Check Core Cooling and direct operators to restart an RCP if required.

C. Complete EOP FR-C.1 in its entirety, then transition to EOP E-1, Loss of Reactor or Secondary Coolant to restart an RCP.

D. Direct operators to go to Appendix B and restart an RCP in parallel with performing the remaining steps of EOP FR-C.1

Proposed Answer: D

Explanation:

A. Incorrect. In accordance with EOP FR-C.1 Appendix B step 4, Continue with the procedure while performing this step.

B. Incorrect. Step 17 does not direct operators to restart a RCP.

C. Incorrect. Step 4 directs operators to go to Appendix B and restart a RCP while continuing the procedure, not to transition to EOP E-1.

D. **Correct.** Step 4 directs operators to go to Appendix B and restart a RCP while continuing the procedure.

Technical References: EOP FR-C.1, Response to Inadequate Core Cooling
References to be provided to applicants during exam: None

Learning Objective: Describe the recovery strategy for inadequate core cooling. (41702)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR 43.5 / 45.13	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and interpret the operable control channel as it applies to the Pressurizer Pressure Control Malfunctions.	**Tier #**	1
	Group #	1
	K/A #	027 AA2.18
	Rating	3.4

Question 78

Given:

Unit 1 is at 100% power and a pressurizer pressure channel failure has resulted in:

- PT-456 indication failing low
- PK04-04, OTΔT C-3 Activating
- PK04-06, Protection Channel Activating
- PK02-04, Safeguard Channel Activating

Based on these conditions, the Shift Foreman should direct:

A. PT-455/474 be selected on the Pressure Channel Selector Switch in accordance with AP-5, Malfunction of Eagle 21 Protection or Control Channel.

B. PT-457/474 be selected on the Pressure Channel Selector Switch in accordance with AP-5, Malfunction of Eagle 21 Protection or Control Channel AP-5.

C. PT-457/474 be selected on the Pressure Channel Selector Switch in accordance with AP-13, Malfunction of Reactor Pressure Control System AP-5.

D. PT-455/474 be selected on the Pressure Channel Selector Switch in accordance with AP-13, Malfunction of Reactor Pressure Control System AP-5.

Proposed Answer: A

Explanation:

A. Correct – AP-5 is entered from all three ARPKs due to failure of the control system due to the channel failure. With PT-456 failed, PT 474 is the only used as a backup channel for PT-455.

B. Incorrect - AP-5 is entered from all three ARPKs due to failure of the control system due to the channel failure. PT 457 as the controlling channel can only have PT-456 as the backup channel and PT-456 is the failed channel.

C. Incorrect – AP-13 would not be entered due to this failure would not result in a plant transient. PT 457 as the controlling channel can only have PT-456 as the backup channel and PT-456 is the failed channel.

D. Incorrect - AP-13 would not be entered due to this failure would not result in a plant transient. With PT-456 failed, PT 474 is the only used as a backup channel for PT-455.

Technical References: OIM 4-4a, Pressurizer Pressure Channel Functions
OIM 4-4b, Pressurizer Pressure Channel Failures
OP AP-5, malfunction of Eagle 21 Protection or Control Channel, Revision 28B
OP AP-13 malfunction of Reactor Pressure Control System, Revision 5A

LA-4A, Pressurizer, Pressure and Level Control

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to determine operability and/or availability of safety related equipment. Steam Gen. Tube Rupture / 3	Tier #	1
	Group #	1
	K/A #	038 2.2.37
	Rating	4.6

Question 79

Unit 1 has tripped and safety injection has actuated.

- An upward trend is observed on RM-15, SJAE
- PK11-18, MAIN STM LINE HI RAD alarm is in
- SG 1-2 level is rising
- All SG pressures are stable
- RCS temperature is 545ºF
- RCS wide range pressure is 1945 psig
- Containment Pressure is 1.2 psig and stable
- EOP E-0, Reactor Trip or Safety Injection, is the procedure in effect

The STA checks SG 1-2 steam flow/feed flow mismatch and quickly estimates RCS leakage to be approximately 7 gallons per hour.

Given the above plant conditions, what procedure should the crew transition to?

A. EOP ECA-3.1, SGTR Loss of Reactor Coolant with Subcooled Recovery

B. EOP E-1, Loss of Reactor or Secondary Coolant

C. EOP E-3, Steam Generator Tube Rupture

D. EOP E-1.2, Post LOCA Cooldown and Depressurization

Proposed Answer: C

Explanation:

A. Incorrect. EOP E-0 does not have a direct transition to EOP ECA 3.1.

B. Incorrect. Plant indications do not require a transition to EOP E-1.

C. Correct. Plant indications require operators to transition to EOP E-3 at step 10 of EOP E-0. Plant indications are for a SG Tube Rupture.

D. Incorrect. Plant conditions do not require entry into EOP E-1.2.

Technical References: EOP E-0, Rx Trip or Safety Injection; EOP ECA-3.1, SGTR Loss of Reactor Coolant with Subcooled Recovery; EOP E-1, Loss of Reactor or Secondary Coolant; EOP E-3, Steam Generator Tube Rupture; EOP E-1.2, Post LOCA Cooldown and Depressurization.

References to be provided to applicants during exam: None

Learning Objective: Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event. (3552)
Apply Technical Specification Bases. (9694,L,M,N)

Question Source: Bank #

(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and interpret DC loads that are lost and the impact on the ability to operate and monitor plant systems.	**Tier #**	1
	Group #	1
	K/A #	058 AA2.03
	Rating	3.9

Question 80

Given:

Unit 2 is at 100% power.

The Shift Foreman observes:

- Main annunciator system has received numerous seemingly unrelated alarms
- Feed flow indicates zero on all steam generators
- Main Feed pump #1 is tripped and Main Feed pump #2 is running
- 4KV Bus F indicated voltage is lowering then goes to zero

Based on these conditions, the Shift Foreman should enter:

A. OP AP-4, Loss of Vital or Nonvital Instrument AC

B. OP AP-15, Loss of Feedwater Flow

C. OP AP-23, Loss of Vital DC

D. OP AP-27, Loss of 4KV and/or 480V Vital Bus

Proposed Answer: **C**

Explanation:

A. Incorrect – Breaker control is DC not AC, but, this will result in multiple alarms.

B. Incorrect – Loss of feedwater is due to closure of the feed regulating valves due to loss of DC power to the train A solenoids.

C. Correct – Loss of vital DC power will result in the failure of the feed regulating valve, the failure of the 4KV bus to transfer due to loss of DC power to the breakers.

D. Incorrect – loss of the 4KV bus resulted from the loss of vital DC. AP-27 is used to recover from a loss of AC power.

Technical References:
References to be provided to applicants during exam: None

Learning Objective: 3478 – Give initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Knowledge of how abnormal operating procedures are used in conjunction with EOPs. Reactor Trip	**Tier #**	1
	Group #	1
	K/A #	EPE 007 2.4.8
	Rating	4.5

Question 81

Unit 1 has tripped and the actions of E-0, Reactor Trip or Safety Injection steps 1 through 4 have just been completed.

- Safety Injection is not required
- RCS Temperature is 535ºF and slowly lowering
- All Steam Dumps, Steam Generator (SG) Blowdown Isolation valves are closed; MSR are reset; and all the MSIVs and MSIV Bypass valves are closed
- All SG Water Levels are 32% and slowly rising
- All SG pressures are 1000 psig and steady

Based on these conditions, the Shift Foreman should enter:

A. E-1.1, SI Termination, then direct operators to Go To E-1.2, Post LOCA Cooldown and Depressurization. In accordance with OP1.DC10, Conduct of Operations, the words "Go To" means to exit E-1.1 and enter E-1.2.

B. E-1.1, SI Termination, then direct operators to Go To E-1.2, Post LOCA Cooldown and Depressurization. In accordance with OP1.DC10, Conduct of Operations, the words "Go To" means to perform E-1.1 parallel with E-1.2.

C. E-0.1, Reactor Trip Response, then direct operators to "implement" OP AP-6, Emergency Boration, and continue to step 2 of E-0.1. In accordance with OP1.DC10, Conduct of Operations, the word "implement" means to perform OP AP-6 after completing the immediate actions of E-0.1.

D. E-0.1, Reactor Trip Response, then direct operators to "implement" OP AP-6, Emergency Boration, and continue to step 2 of E-0.1. In accordance with OP1.DC10, Conduct of Operations, the word "implement" means to perform OP-AP-6 in parallel with E-0.1.

Proposed Answer: D

Explanation:

A. Incorrect. For the given plant conditions enter E 0.1 (not E-1.1) and direct operators to "implement" OP AP-6 Emergency Boration and continue to step 2. In accordance with OP1.DC10, Conduct of Operations, the words Go To do mean to exit the procedure in effect and enter the directed procedure. Step 4 of E-0.0, the Response Not Obtained column directs operators to enter E-0.1 for the given plant conditions.

B. Incorrect. EOP E0.1 should be entered, not E-1.1. Also, OP AP-6 (not E-1.2) should be entered in parallel with the procedure in effect. This is the definition of "implement" not "Go To."

C. Incorrect. E-0.1 is correct. This is also the correct action in accordance with OP AP-6 but EOP E0.1 should still be executed in parallel. Answer C has an incorrect definition of "implement."

D. **Correct.** For the given plant conditions enter E 0.1 and direct operators to "implement" OP AP-6 Emergency Boration and continue to step 2. In accordance with OP1.DC10, Conduct of Operations, the word "implement" means to perform the second procedure in parallel with the procedure in effect.

Technical References: E-0, Reactor Trip or Safety Injection; OP1.DC10, Conduct of Operations; E-0.1, Reactor Trip Response.

References to be provided to applicants during exam: None

Learning Objective: Describe what procedure or procedure set would be used in an emergency event, based on plant mode/conditions. (6764)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 43.5	

Examination Outline Cross-Reference	Level	SRO
Knowledge of the Emergency Boration system set points, interlocks and automatic actions associated with EOP entry conditions	**Tier #**	1
	Group #	2
	K/A #	000024 2.4.2
	Rating	4.6

Question 82

Given:

Unit 1 is experiencing an uncontrolled cooldown following a reactor trip with no ESF actuation.

The Shift Foreman should enter EOP _____ and ensure the boric acid flow rate is at least _____ gpm per OP AP-6, Emergency Boration.

A. E-0, Reactor Trip or Safety Injection; 30

B. E-0, Reactor Trip or Safety Injection; 50

C. E-0.1, Reactor Trip Response; 30

D. E-0.1, Reactor Trip Response; 50

Proposed Answer: C

Explanation:

A. Incorrect – EOP E-0 does not direct entry into OP AP-6 to emergency borate the plant. OP AP-6 Step 1 verifies at least 30 GPM of Boric Acid Flow.

B. Incorrect – EOP E-0 does not direct entry into OP AP-6 to emergency borate the plant. OP AP-6 Step 1 verifies at least 30 GPM of Boric Acid Flow instead of the 50 GPM specified in the answer. The 50 GPM emergency boration flowrate is the flow in which the Emergency Boration Flowmeter FI-113 pegs high specified in the Note on page 3 of OP AP-6.

C. Correct – EOP E-0.1 requires entry into OP AP-6 due to uncontrolled cooldown per the Step 1 ROIs. OP AP-6 Step 1 verifies at least 30 GPM of Boric Acid Flow.

D. Incorrect – EOP E-0.1 requires entry into OP AP-6 due to uncontrolled cooldown per the Step 1 ROIs. OP AP-6 Step 1 verifies at least 30 GPM of Boric Acid Flow instead of the 50 GPM specified in the answer. The 50 GPM emergency boration flowrate is the flow in which the Emergency Boration Flowmeter FI-113 pegs high specified in the Note on page 3 of OP AP-6.

Technical References: EOP E-0, Reactor Trip and Safety Injection, Revision 40
EOP E-0.1, Reactor Trip Response, Revision 35
OP AP-6, Emergency Boration, Revision 19

References to be provided to applicants during exam: None

Learning Objective:

Question Source: Bank #
(note changes; attach parent) Modified Bank #
New X

Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.45.7-8	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: AA2.01 ARM system indications	**Tier #**	1
	Group #	2
	K/A #	APE 036 AA2.01
	Rating	3.9

Question 83

Unit 1 is in a refueling outage with the conditions:

- Alarm PK 11-10 FHB HIGH RADIATION RE-58/59 comes in
- Alarm PK 02-06 CONTMT VENT ISOLATION comes in
- The Spent Fuel Pool level is 25 feet above spent fuel assemblies and steady

The Shift Foreman should enter:

A. OP AP-21 Irradiated Fuel Damage, the Fuel Handling Building Evacuation Horn is sounding, and the Fuel Handling Building Ventilation is operating in the iodine removal mode.

B. OP AP-22 Spent Fuel Pool Anomalies, the Fuel Handling Building Evacuation Alarm is sounding, and Containment Ventilation has isolated.

C. AD8.DC54 Containment Closure, the Containment Evacuation Alarm is sounding, and the Fuel Handling Building Ventilation is operating in the iodine removal mode.

D. OP AP-21 Irradiated Fuel Damage, the Containment Evacuation Alarm is sounding, and the Fuel Handling Building Ventilation is operating in the iodine removal mode.

Proposed Answer: A

Explanation:

A. Correct**.** OP AP-21 is correct. RE-58/59 automatically sound the FHB Evacuation Horn, and shift the FHB Ventilation to Iodine Removal mode.

B. Incorrect. OP AP-22 is not correct since SFP level meets the minimum level and is not changing. Also, the Containment Evacuation Alarm is not sounded until Containment Radiation alarms come in and are verified by the Refueling SRO or SFM. Containment only isolates if the accident is inside containment; plant conditions are for an accident in the FHB.

C. Incorrect. Plant conditions are not met for entering AD8.DC54, which is for an accident inside of Containment. Also, the Containment Evacuation Alarm is not sounded until Containment Radiation alarms come in and are verified by the Refueling SRO or SFM.

D. Incorrect. The Containment Evacuation Alarm is not sounded until Containment Radiation alarms come in and are verified by the Refueling SRO or SFM. The FHB Evacuation Horn should be sounding due to RE-58.59 alarming.

Technical References: Operations Lesson Radiation Monitoring; OP AP-21 Irradiated Fuel Damage; OP AP-22 Spent Fuel Pool Anomaly; AD8.DC54 Containment Closure.

References to be provided to applicants during exam: None

Learning Objective: Describe controls, indications, and alarms associated with the Radiation

Monitoring System. (37875)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 43.5	

Examination Outline Cross-Reference	Level	SRO
Knowledge of the emergency action level thresholds and classifications associated with Accidental Gaseous Radwaste Release.	**Tier #**	1
	Group #	2
	K/A #	000060 2.4.41
	Rating	4.6

Question 84

An Unusual Event would have to be declared per the Emergency Plan if an unplanned release of gaseous radioactivity occurs that is three times the limit contained in _________ for greater than 60 minutes.

A. The DCPP Technical Specifications

B. The Radiological Effluent Technical Specifications

C. The Final Safety Analysis Report

D. 10 CFR 20; Standards for Protection Against Radiation

Proposed Answer: B

Explanation:

A. Incorrect – The DCPP Technical Specifications do not contain gaseous radioactivity release limits.

B. Correct – EAL R1 states that the limit is based on the values found in the Radiological Effluent Technical Specifications.

C. Incorrect – The FSAR does not contain gaseous radioactivity release limits.

D. Incorrect 10 CFR 20 does not contain gaseous radioactivity release limits

Technical References: EP G-1 Emergency Action Level Matrix, Revision 40

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Whether an alarm channel is functioning properly	**Tier #**	1
	Group #	2
	K/A #	061 AA2.04
	Rating	3.5

Question 85

Unit 1 is in Mode 4.

While performing a channel check of RM-11, Containment Activity, an operator notes the current reading is 2.5 times what is was 12 hours ago on the last channel check. RM-12, Containment Activity, indicates approximately the same value as it did 12 hours ago.

Based on this, Technical Specification 3.4.15, RCS Leakage Detection Instrumentation (attached):

A. Is still met and requires no action.

B. Requires grab samples for the containment atmosphere be analyzed every 24 hours.

C. Requires restoration of RM-11 to an operable status within 30 days.

D. Requires Unit 1 be in Mode 3 within 6 hours.

Proposed Answer: A

Explanation:

A. Correct. Because the second reading is more than twice the previous reading, RM-11 must be declared inoperable. However, per the provided LCO, only one containment atmosphere monitor is required. Since RM-12 is still operable, the LCO is met and no action is required.

B. Incorrect. This action is required if both instruments are inoperable.

C. Incorrect. This action is required if the containment sump monitor is inoperable.

D. Incorrect. This action is required if a required action in this technical specification is not met.

Technical References: Technical Specification 3.4.15.

References to be provided to applicants during exam: TS 3.4.15 and Page 18 of STP I-1A (Unit1) Attachment 12.1.

Learning Objective: Discuss significant Technical Specifications and Equipment Control Guidelines associated with the Radiation Monitoring System. (9694, 9697, 9633)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 43.5	

Examination Outline Cross-Reference	Level	SRO
Knowledge of EOP entry conditions and immediate actions steps associated with the Chemical and Volume Control system.	**Tier #**	2
	Group #	1
	K/A #	000004 2.4.1
	Rating	4.8

Question 86

EOP E-0, Reactor Trip or Safety Injection, actions are in progress on Unit 1 following a Reactor Trip with a Safety Injection actuation.

Step 18, Control Charging flow to Maintain PZR Level, requires entry into _______ if ________ fails to maintain PZR level between 17% and 60%.

A. EOP E-1, Loss of Reactor or Secondary Coolant; starting an additional ECCS CCP

B. EOP E-1, Loss of Reactor or Secondary Coolant; adjusting FCV-128 and HCV-142

C. EOP E-1.2, Post LOCA Cooldown and Depressurization; starting an additional ECCS CCP

D. EOP E-1.2, Post LOCA Cooldown and Depressurization; adjusting FCV-128 and HCV-142

Proposed Answer: D

Explanation:

A. Incorrect – Step 18 directs entry into EOP E-1.2 if pressurizer level is less than 17% and lowering after attempting to maintain level by adjusting FCV-128 and HCV-142. Step 14 stops all but one CCP with Step 16 isolating charging injection.

B. Incorrect - Step 18 directs entry into EOP E-1.2 if pressurizer level is less than 17% and lowering after attempting to maintain level by adjusting FCV-128 and HCV-142. Step 18 adjusts FCV-128 and HCV-142 to maintain pressurizer level.

C. Incorrect – The RNO for step 18 if adjustment to FCV-128 and HCV-142 cannot maintain pressurizer level between 17% and 60% requires the realignment of charging injection and entry into EOP E-1.2. Step 14 stops all but one CCP with Step 16 isolating charging injection.

D. Correct - Step 18 directs entry into EOP E-1.2 if pressurizer level is less than 17% and lowering after attempting to maintain level by adjusting FCV-128 and HCV-142.

Technical References: EOP E-0, Reactor Trip or Safety Injection, Revision 40
EOP E-1, Loss of Reactor or Secondary Coolant, Revision 30
EOP E-1.2, Post LOCA Cooldown and Depressurization, Revision 20A
Lesson Guide LPE-0, Reactor Trip or Safety Injection, Revision 11

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X

	Comprehensive/Analysis
10CFR Part 55 Content:	55.43.5

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: erratic power supply operation	**Tier #**	1
	Group #	2
	K/A #	012 A2.04
	Rating	3.2

Question 87

Unit 1 is shutting down for a maintenance outage and:

- Reactor Power is ≤ 10E-8 amps
- Intermediate Range Nuclear Instrument Channel 35 loses compensating voltage due to a power supply malfunction resulting in erratic voltage.

Technical Specification 3.3.1, Reactor Trip System Instrumentation, requires _______. The bases for having the Intermediate Range Nuclear Instruments operable is to mitigate the consequences of ________.

A. Reactor power be less than P-6 or greater than P-10 within 24 hours; a dilution accident.

B. Placing the channel in trip within 72 hours; an RCCA rod bank withdrawal accident.

C. Placing the channel in trip within 72 hours; a dilution accident.

D. Reactor power be less than P-6 or greater than P-10 within 24 hours; an RCCA rod bank withdrawal accident.

Proposed Answer: D

Explanation:

A. Incorrect. The bases for this TS is to mitigate the consequences of a rod withdrawal accident.

B. Incorrect. The correct TS action is to have reactor power less than P-6 or greater than P-10 within 24 hours.

C. Incorrect. The bases for this TS is to mitigate the consequences of a rod withdrawal accident. Second, the action is incorect.

D. Correct. Per TS 3.3.1, reactor power must be less than P-6 or greater than P-10 within 24 hours; The TS bases document states this is to mitigate the consequences of an RCCA rod bank withdrawal accident.

Technical References: TS 3.3 and Bases TS 3.3; Systems Lesson Reactor Protection System LB-6A; ARPK 03-06
References to be provided to applicants during exam: TS 3.3.1

Learning Objective: Discuss significant Technical Specifications and Equipment Control Guidelines associated with the Reactor Protection System. (9694, 9697, 9633)

- Apply TS 3.3 Technical Specification LCOs. (9697C)
- Apply TS 3.3 Technical Specification bases (SROs only). (9694C)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of a stuck open PORV or code safety on the Pressurizer Relief Tank; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of a stuck open PORV or code safety.	**Tier #**	2
	Group #	1
	K/A #	007 A2.01
	Rating	4.2

Question 88

GIVEN:

- Unit 1 is at 100% power
- PK05-20, PZR Relief/Safety Valve Open – ON
- PK05-23, PZR Safety or Relief Line Temp – ON
- Pressurizer pressure dropped rapidly
- SI has not actuated

The Shift Foreman should enter ________ and at 10 psig, __________ will close if open.

A. AP-13, Malfunction of Reactor Pressure Control; PCV-472, PRT Vent Header Pressure Control valve

B. AP-1, Excessive Reactor Coolant System Leakage; RCS-8034A/B, Gas Analyzer valves

C. E-1, Loss of Reactor Coolant or Secondary Coolant; PCV-472, PRT Vent Header Pressure Control valve

D. AP-13, Malfunction of Reactor Pressure Control; RCS-8034A/B, Gas Analyzer valves

Proposed Answer: A

Explanation:

A. Correct – PK05-23 directs going to AR PK05-20 if PK05-20 and PK05-23 are in alarm. AR PK05-20 directs entry into AP-13 due to pressurizer pressure has dropped rapidly. At 10 psig in the PRT PCV-472 will close if the valve is in the open position to isolate the PRT from the vent header.

B. Incorrect – AP-1 has potential symptoms as given above but is not directed to be entered from the indication given. RCS-8034A/B will close on a Phase A containment isolation not on PRT pressure.

C. Incorrect – E-1 entry is post SI Actuation therefore E-1 would not be entered given this information. At 10 psig in the PRT PCV-472 will close if the valve is in the open position to isolate the PRT from the vent header.

D. Incorrect - PK05-23 directs going to AR PK05-20 if PK05-20 and PK05-23 are in alarm. AR PK05-20 directs entry into AP-13 due to pressurizer pressure has dropped rapidly. RCS-8034A/B will close on a Phase A containment isolation not on PRT pressure.

Technical References: AR PK05-20, PZR Relief/Safety Valves Open, Revision
AR PK05-23, PZR Safety or Relief Line Temp, Revision 21
OP AP-1, Excessive Reactor Coolant System Leakage, Revision 20
OP AP-13, Malfunction of Reactor Pressure Control System, Revision 5A

EOP E-1, Loss of Reactor or Secondary Coolant, Revision 30

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to perform specific system and integrated plant procedures during all modes of plant operation. Service Water	**Tier #**	1
	Group #	2
	K/A #	076 2.1.23
	Rating	4.4

Question 89

GIVEN:

- Unit 1 is at 90% power
- Annunciator BAR/RACKS SCREENS, PK13-01, is lit
- An equipment operator calls the control room and reports heavy kelp loading on the Circulating Water System traveling screens
- Unit 1 Screen differential pressure is 72 inches and slowly rising
- Unit 2 is in a maintenance outage; both trains of Service Water are out of service for corrective maintenance.
- Circulating Water Pump 1-1 is in service. CWP 1-2 is tagged out for maintenance.

Which of the following are the correct actions to take for given plant conditions?

A. Enter OP AP-25, Rapid Load Reduction.
2. Reduce power to <50%.
3. Enter OP F-1:VI Service Cooling Water – Alternate Cooling, to align the Screen Wash System to cool the SCW HX

B. Enter OP AP-7, Degraded Condenser.
2. Reduce power to <50%.
3. Enter OP O-23 Operating Order 23 – Alignment Checklist

C. Enter OP AP-25, Rapid Load Reduction
2. Trip the Reactor and enter E-0.
3. Enter OP F-1:I Service Cooling Water – Make Available

D. Enter OP AP-7, Degraded Condenser.
2. Trip the Reactor and enter E-0.
3. Enter OP F-1:VI Service Cooling Water – Alternate Cooling, to align the Screen Wash System to cool the SCW HX.

Proposed Answer: D May have to replace this question.

Explanation:

A. Incorrect. 1. Enter OP AP-7, Degraded Condenser should be entered, not OP AP-25. 2. While entering OP AP-25 is a judgment call, plant conditions call for OP AP- 7 which directs operators to trip the reactor vice reducing power because only 1 CWP is operating and reactor power is above 25%.

B. Incorrect. 2. OP AP-7 directs operators to trip the reactor vice reducing power.

C. Incorrect. 1. Enter OP AP-7, Degraded Condenser should be entered, not OP AP-25. 3. OP F-1:VI should be entered to provide alternate cooling water to the SCW HX

D. Correct. These are the correct actions for plant conditions.

Technical References: OP AP-25, Rapid Load Reduction; OP AP-7, Degraded Condenser; OP F-1:VI Service Cooling Water – Alternate Cooling; Operations

Lesson Service Water System

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the Service Cooling Water System. (37109)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 41.10 / 43.5 / 45.2 / 45.6	

<table>
<tr><td>Examination Outline Cross-Reference</td><td>Level</td><td>SRO</td></tr>
<tr><td rowspan="5">Ability to (a) predict the impacts of a Containment Evacuation (including recognition of the alarm) on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of a Containment Evacuation.</td><td>Tier #</td><td>2</td></tr>
<tr><td>Group #</td><td>1</td></tr>
<tr><td>K/A #</td><td>000103
A2.04</td></tr>
<tr><td>Rating</td><td>3.6</td></tr>
</table>

Question 90

Unit 1 is Off-loading Fuel

While in process of withdrawing a fuel assembly the Red overload light on the Manipulator Crane Control Console illuminated and the refueling crew observed a release of gas bubbles from the fuel assembly.

Given the following indications:

- PK02-06, Containment Vent Isolation – Alarm off
- PK11-19, Containment Radiation – Alarm off
- PK11-21, High Radiation – Alarm on

The alarm is a(n) __________ and the Shift Foreman should ensure the ________.

A. electronic warbler that sounds a series of falling tones and red flashing lights; Containment Purge Exhaust is in service per OP H-4:I, Containment Ventilation Make Available and Place in Service

B. constant, high-pitched squeal and red flashing lights; Containment Purge Exhaust is in service per OP H-4:I, Containment Ventilation Make Available and Place in Service

C. is an electronic warbler that sounds a series of falling tones and red flashing lights; Containment Purge Exhaust is isolated per OP AP-21, Irradiated Fuel Damage

D. is a constant, high-pitched squeal and red flashing lights; Containment Purge Exhaust is in isolated per OP AP-21, Irradiated Fuel Damage

Proposed Answer: C

Explanation:

A. Incorrect – Correct alarm but OP AP-21 requires that containment purge exhaust is verified closed vice in-service during a containment evacuation. OP H-4:I does give direction for placing the containment purge system in service but this condition requires the system to be isolated.

B. Incorrect – Incorrect alarm and OP AP-21 requires that containment purge exhaust is verified closed vice in-service during a containment evacuation. OP H-4:I does give direction for placing the containment purge system in service but this condition requires the system to be isolated.

C. Correct – Correct alarm and OP AP-21 requires that containment purge exhaust is verified closed.

D. Incorrect - Incorrect alarm and OP AP-21 requires that containment purge exhaust is verified closed.

Technical References: OP AP-21, Irradiated Fuel Damage, Revision 10
References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of forced circulation of Reactor Coolant	**Tier #**	2
	Group #	2
	K/A #	002 A2.03
	Rating	4.1

Question 91

GIVEN:

- Unit 1 is at 100% power
- PK 5-01, RCP 1-1, is IN for input 505 RCP MOTOR OIL LEVEL HI LO
- PK 5-05, RCP VIBRATION alarm is IN and verified to be VALID
- Differential Pressure across RCP 2-1 Number 1 seal is 260 psid and steady
- The Plant Parameters Computer shows that RCP 1-1 Top Radial Bearing Temp is 201ºF and steady

The Shift Foreman should __________.

A. Trip the reactor and enter E-0 Reactor Trip or Safety Injection because RCP 1-1 Motor Oil Cooler CCW temperature is too low; Trip RCP 1-1, and close the associated PZR Spray Valve.

B. Trip the reactor and enter E-0 Reactor Trip or Safety Injection because the RCP 1-1 Motor Oil Cooler CCW temperature is too high; then Trip RCP 1-1

C. Check if No.1 and No. 2 seal status allow continued RCP operation because RCS pressure is low

D. Check if No.1 and No. 2 seal status allow continued RCP operation because the seal injection temperature is high

Proposed Answer: A

Explanation:

A. Correct. 1) IAW AOP-28, RCP Malfunction Section A, these actions should be performed - as indicated by both PK 5-01 and 5-05 having alarmed. 2) Probable causes include RCP Motor Oil Cooler CCW temperature too LOW but not Low RCS pressure or High Seal Injection Temperature.

B. Incorrect. RCP oil temperature is not too high.

C. Incorrect. 1) These would be the correct actions for a No. 1 seal failure; which would be indicated by abnormal differential pressure across RCP seal number 1. 2) Low RCS pressure would not cause PK 5-01.

D. Incorrect. 1) No.1 seal differential pressure is normal, therefore, this is the wrong action to take. 2). High Seal Injection Temperature would not cause PK 5-01.

Technical References: Operations Lesson LAR-1 RCP Malfunctions, AOP-28 RCP Malfunction
References to be provided to applicants during exam: None

Learning Objective: Explain the causes of RCP abnormal conditions. (6104) Given an abnormal condition, summarize the major actions of the OP AP-28 to mitigate an event in progress. (3477)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 43.5	

Examination Outline Cross-Reference	Level	SRO
Knowledge of the Steam Generator system parameters and logic used to assess the status of the safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	**Tier #**	2
	Group #	2
	K/A #	000035 2.4.21
	Rating	4.6

Question 92

Unit 1 Containment is NOT in an Adverse Condition, total AFW Flow is 400 GPM, and the Steam Generators indicate:

Steam Generator	Water Level	Pressure
1-1	18%	1050 PSIG
1-2	14%	1090 PSIG
1-3	19%	1040 PSIG
1-4	22%	1035 PSIG

In accordance with EOP F-0, Critical Safety Function Status Trees (CSFST), the given indications will result in entry into _______________.

A. FR-H.1, Response to Loss of Secondary Heat Sink

B. FR-H.2, Response to Steam Generator Over pressurization

C. FR-H.4, Response to Loss of Normal Steam Release Capacities

D. FR-H.5, Response to Steam Generator Low Level

Proposed Answer: C

Explanation:

A. Incorrect – All Steam Generator Levels would have to be < 15%. Only Steam Generator 1-2 is < 15%.

B. Incorrect – All Steam Generators are < 1115 PSIG.

C. Correct - Steam Generator 1-2 pressure is > 1065 PSIG therefore a Yellow path to FR-H.4 would be entered.

D. Incorrect – All Steam Generators are not > 15% which would allow a yellow path to FR-H.5 but the prior condition was not met and would not have resulted in continuing the flowpath to this question.

Technical References: LPE-H, Secondary Heat Sink Challenge Response, Revision 13

EOP FR-0, Critical Safety Function Status Trees, Revision 17

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. Station Air	**Tier #**	2
	Group #	2
	K/A #	079 2.4.4
	Rating	4.7

Question 93

GIVEN:

- Unit 1 is at 100% power.
- T_{avg} is 560ºF
- PZR Level is 55%
- PZR Temperature is slightly above normal and slowly rising
- TCV-130, Letdown HX CCW outlet is Open
- FCV-110B and 111B, Blender outlet FCVs are Closed
- PI-380, Instrument Air Header Pressure reads 85 psig and is slowly lowering
- MFW Pump and Condensate Pump recirculation FCVs are Closed
- Feed Water Regulating and Bypass Valves are Closed
- Steam Generator Water Levels are lowering rapidly

The Shift Foreman should enter:

A. OP AP-5, Malfunction of Protection or Control Channel.

B. OP AP-9, Loss of Instrument Air.

C. OP AP-13, Malfunction of Reactor Pressure Control System.

D. OP AP-15, Loss of Feedwater Flow, and reduce reactor power to less than 80 percent.

Proposed Answer: B

Explanation:

A. Incorrect.

B. Correct. From the given indications the correct procedure to enter is a loss of instrument air.

C. Incorrect.

D. Incorrect.

Technical References: OP AP-9, Loss of Instrument Air; EOP E-0, Reactor Trip or Safety Injection; OP AP-15, Loss of Feedwater Flow; Operations Lesson LPA-9, Loss of Instrument Air.

References to be provided to applicants during exam: None

Learning Objective: Given initial conditions, assumptions, and symptoms, determine the correct abnormal operation procedure to be used to mitigate an operational event. (3478) List the effects that a loss of Instrument Air would have on the plant. (3541)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 43.2 / 45.6	

Examination Outline Cross-Reference	Level	SRO
Knowledge of procedures and limitations involved in core alterations	**Tier #**	3
	Group #	-
	K/A #	2.1.36
	Rating	4.1

Question 94

Unit 1 is currently performing core loading in accordance with OP B-8DS2, Core Loading.

Which of the following would result in the suspension of core loading?

A. An unexpected increase in count rate on one responding nuclear channel by a factor of 2.

B. A change in Reactor Coolant System temperature of greater than 15°F.

C. A change in boron concentration from the nominal value at the start of core loading by ±30 ppm.

D. An unexpected increase in count rate on two responding nuclear channels by a factor of 3.

Proposed Answer: D

Explanation:

A. Incorrect – OP B-8DS2 specifies that an unexpected increase of one responding nuclear channel by a factor of 3 will result in a suspension of core loading.

B. Incorrect - OP B-8DS2 specifies a change in Reactor Coolant System temperature of greater than 20°F will result in a suspension of core loading.\

C. Incorrect - OP B-8DS2 specifies a change in boron concentration from the nominal value at the start of core loading by ±50 ppm will result in a suspension of core loading.

D. Correct - OP B-8DS2 specifies that an unexpected increase of one responding nuclear channel by a factor of 3 will result in a suspension of core loading. Therefore, two responding channels with unexpected increase in count rate by a factor of three would result in a suspension of core loading.

Technical References: OP B-8DS2, Core Loading, Revision 50
References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.6	

Examination Outline Cross-Reference	Level	SRO
Ability to identify and interpret diverse indications to validate the response of another indication. Conduct of Operations	Tier #	3
	Group #	-
	K/A #	2.1.45
	Rating	4.3

Question 95

Given:

- Unit 1 is at 100% power
- Tave is 577ºF
- LT-459 indicates 60%, LT-460 indicates 53%, LT-461 indicates 56%.
- PT-455 indicates 2235 psig, PT-456 indicates 2100 psig, PT-457 indicates 2215 psig.
- RM-3, Oily Water Separator, needle is rock steady and pegged low.
- RM-11, Containment Air Particulate Detector, FILTER NOT IN MOTION light is OUT.

Based on these indications;

A. LT-460 and PT-456 are inoperable

B. LT-460 and RM-11 are inoperable

C. PT-456 and RM-3 are inoperable

D. LT-460, PT-456, RM-3, and RM-11 are all inoperable

Proposed Answer: C

Explanation:

A. Incorrect. PT-456 is operable because Pressurizer Level channels may be 8.5% different.
B. Incorrect. RM-11 is operable; this is the expected condition for channel checks of Westinghouse radiation monitors.
C. Correct. PT-456 is greater than 5% different from the other channels and RM-3 should have slight needle oscillations.
D. Incorrect. See above.

Technical References: Channel Check Criteria B-5 Rev. 5
References to be provided to applicants during exam: None
Learning Objective: To be determined.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 43.5 / 45.4	

Examination Outline Cross-Reference	Level	SRO
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	**Tier #**	3
	Group #	-
	K/A #	2.2.36
	Rating	4.2

Question 96

Unit 1 is operating at 100% power and:

- TDAFW Pump 1-1 is inoperable due to maintenance
- Twelve hours later the Emergency Diesel Generator 1-3 is declared inoperable due failure of the Monthly Surveillance test.

In order to be in compliance with the Technical Specifications, the maximum time allowed to be in Mode 3 is ____________ hours.

A. 6

B. 10

C. 66

D. 70

Proposed Answer: B

Explanation:

A. Incorrect – This is a direct entry into TS 3.7.5 C.1 due to Two AFW Trains inoperable in Modes 1, 2 or 3. The second train does not become inoperable until 4 hours after the EDG is declared inoperable according to TS 3.8.1 B.2.

B. Correct – This includes the 4 hours from TS 3.8.1 B.2 and the 6 hours from TS 3.7.5 C.1

C. Incorrect – This includes the 60 hours left on TS 3.7.5 B.1 plus the 6 hours from TS 3.7.5 C.1. TS 3.7.5 C.1 is entered after 4 hours have expired after declaring the EDG inoperable (TS 3.8.1 B.2).

D. Incorrect – This includes the 60 hours left on TS 3.7.5 B.1 plus the 6 hours from TS 3.7.5 C.1 and the 4 hours from TS 3.8.1 B.2.

Technical References: Technical Specifications 3.7.5 and 3.8.1
References to be provided to applicants during exam: Technical Specifications 3.7.5 and 3.8.1
Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Ability to comply with radiation work permit requirements during normal or abnormal conditions. Radiation Control	Tier #	3
	Group #	-
	K/A #	2.3.7
	Rating	3.6

Question 97

The Unit 1 Shift Manager has just declared an Alert following a design basis Loss of Coolant Accident. The Emergency Operations Facility and the Technical Support Center have not yet been activated; no turnovers have occurred.

For the purpose of Radiological Assessment Sampling, workers will be exposed to radiation levels in excess of the 10 CFR Part 20 exposure limits.

In the current situation, the ____________ the authority to authorize the dose on an EP RB-2, Attachment 9.7, Emergency Exposure Permit.

A. Shift Manager ONLY has

B. Shift Manager OR the Site Emergency Coordinator have

C. Site Emergency Coordinator OR the Radiation Protection Manager have

D. Radiation Protection Manager OR the Plant Manager

Proposed Answer: A Check on titles

Explanation:

A. Correct. Only the Emergency Director or in this case, the Shift Manager (acting ED) has the has the unilateral authority and non delegable responsibility for authorizing an individual emergency worker to exceed normal 10 CFR 20 exposure limits.
B. Incorrect. The TSC is not yet manned, the SEC is not yet stationed.
C. Correct. The Shift Manger has Command and Control of the emergency response for the given plant conditions and until a turnover happens with the SEC or ED.
D. Incorrect. As explained above.

Technical References: RCP-D-201, RWP Writing Guide; OP AP-31, Rapid Containment Entry; Operations Lesson LEP 3 EP RB Procedures.

References to be provided to applicants during exam: None

Learning Objective: State who may authorize emergency doses. (9848)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	CFR: 41.12 / 45.10	

Examination Outline Cross-Reference	Level	SRO
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.	**Tier #**	3
	Group #	-
	K/A #	2.3.12
	Rating	3.7

Question 98

Unit 1 is 100%

A non-emergency containment entry is being performed in accordance with RCP D-230, Radiological Control for Containment Entry.

The Moveable Incore Detector keys are maintained in the possession of _______ during all containment entries.

A. Operations Manager

B. Shift Manager

C. Radiological Protection Manager

D. Radiological Protection Foreman

Proposed Answer: D

Explanation:

A. Incorrect - RCP D-230 states that the RP Foreman or designee maintains the MIDS keys during containment entries.

B. Incorrect - RCP D-230 states that the RP Foreman or designee maintains the MIDS keys during containment entries.

C. Incorrect - RCP D-230 states that the RP Foreman or designee maintains the MIDS keys during containment entries.

D. Correct - RCP D-230 states that the RP Foreman or designee maintains the MIDS keys during containment entries.

Technical References: RCP D-230, Radiological Control for Containment Entry, Revision 21
References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.45.9-10	

Examination Outline Cross-Reference	Level	SRO
Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. Emergency Procedures/Plan	Tier #	3
	Group #	-
	K/A #	2.4.22
	Rating	4.4

Question 99

Given the following conditions following a reactor accident:

- RCS has cooled down from 577ºF to 481ºF in the last hour
- RCS pressure is 460 psig
- RCPs are secured
- Source Range (SR) detectors failed to energize
- Intermediate Range Start-up Rate (SUR) is +0.1 Decades Per Minute (DPM)
- All Core Exit Thermocouples indicate approximately 650ºF
- RVLIS indicates 25%
- Containment Pressure is 24 psig and no Containment Spray Pumps are running
- All T_C instruments indicate 225ºF
- All control rods are fully inserted

In accordance with the bases for EOP F-0, Critical Safety Function Status Trees, which Safety Function has the highest priority and why?

A. Core Cooling is under severe challenge (magenta). When multiple Safety Functions are under severe challenge (magenta), the highest priority safety function is restored first; Core Cooling is the highest priority safety function under severe challenge.

B. RCS Integrity is under severe challenge (magenta). The 6 safety functions have an order of priority; RCS Integrity is the highest priority under a severe challenge and is therefore addressed first.

C. RCS Integrity is under severe challenge (magenta). The safety function with the highest severity level is addressed first. The other safety functions are in a Not Satisfied (yellow) condition.

D. Containment is under severe challenge (magenta). The safety function with the highest severity level is addressed first. The other safety functions are in a Not Satisfied (yellow) condition.

Proposed Answer: A

Explanation:

A. Correct. IF no extreme challenges exist, THEN the operator shall stop procedure in effect and initiate functional restoration to restore the highest priority critical safety function under severe challenge. Core Cooling, RCS Integrity, and Containment are under severe challenge (magenta) and Subcriticality is Not Satisfied (yellow). Therefore, Core Cooling is the highest priority safety function at this time.

B. Incorrect. True - RCS Integrity is under severe challenge (magenta). The bases further prioritize the safety functions during accidents based on the level of the challenge to each function. RCS Integrity is not a higher priority than Core Cooling.

C. Incorrect. RCS Integrity is also under severe challenge (magenta) but is a lower priority than another safety function under severe challenge. Core Cooling is also under severe challenge and is higher priority. Containment is also under severe challenge.

D. Incorrect. Core Cooling is under severe challenge (magenta). The second part of the answer is a true statement but it does not have any effect on the priority of the safety functions.

Technical References: EOP F-0, Critical Safety Function Status Trees; Westinghouse Background Document Emergency Response Guidelines F-0 Critical Safety Function Status Trees

References to be provided to applicants during exam: Critical Safety Function Status Trees

Learning Objective: Apply the Rules of Usage in EOPs for the CSFSTs and FRGs, including:

- the six status trees
- the priority of use of the status trees
- the priority of use of the color of each CSF
  when to monitor and/or implement the CSFSTs and FRGs (38107)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	CFR: 43.5 / 45.12	

Examination Outline Cross-Reference	Level	SRO
Knowledge of the lines of authority during implementation of the emergency plan.	**Tier #**	3
	Group #	-
	K/A #	2.4.37
	Rating	4.1

Question 100

The ______________has the authority to downgrade an Emergency Classification Level at the _________________ level.

A. Shift Manager, Notice of Unusual Event

B. Site Emergency Coordinator; Notice of Unusual Event

C. Shift Manager; Alert

D. Site Emergency Coordinator; Alert

Proposed Answer: A

Explanation:

A. Correct – EP G-1 states that the Shift Manager may downgrade an Unusual Event to no ECL.

B. Incorrect - EP G-1 states that the Shift Manager may downgrade an Unusual Event to no ECL. The Site Emergency Coordinator may upgrade an event to a higher ECL until the Emergency Director assumes responsibility in the EOF.

C. Incorrect – EP G-1 states that the Shift Manager and Site Emergency Coordinator shall not downgrade an event classified at the Alert or Higher level at any time.

D. Incorrect - EP G-1 states that the Shift Manager and Site Emergency Coordinator shall not downgrade an event classified at the Alert or Higher level at any time.

Technical References: EP G-1, Emergency Plan and Emergency Plan Activation, Revision 40
LEP-2, Emergency Plan Procedures, Revision 12

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.45.13	