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November 2001

RO Written Exam

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1___
	Group #	__1__	__1___
	K/A #	295006 AA2.02	
	Importance Rating	_4.3_	_4.4_

Proposed Question: 2 / 1

The plant has just experienced an ATWS in which several control rods had their scram inlet and outlet valves fail to open and did not fully insert. The control room supervisor orders that these control rods be manually inserted using RMCS. Based on the following indication on the full core display, which one of the following control rods must be inserted?

A. Control rod 22-27 has the BLUE light ON
B. Control rod 22-35 has a position indication of 00 on the four rod display
C. Control rod 18-31 has the RED light ON
D. Control rod 26-39 has the GREEN light ON

Proposed Answer: C. Control rod 18-31 has the RED light ON

Explanation (Optional):
A. BLUE light lit means that both scram valves opened and full in by stem
B. Reactor will be shutdown under all conditions if rods at 00 or 02, 00 is full in
D. GREEN light ON means rod is full inserted
C. RED indication is full withdrawn.

Technical Reference(s): SDLP-03F, Reactor Manual Control; EP-3, Backup Manual Control Rod Insertion

Proposed references to be provided to applicants during examination: None

Learning Objective: Ability to determine and/or interpret the following as they apply to SCRAM: (CFR: 41.10 / 43.5 / 45.13) AA2.02 Control rod position 4.3*/ 4.4*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41 __10_
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1___
Group #	__1__	__1___
K/A #	295007 2.3.4	
Importance Rating	_2.5_	_3.1_

Proposed Question: 3 / 2

A large break loss of coolant accident (LOCA) has occurred. All low pressure ECCS systems started except the "A" & "C" RHR pumps. that were out of service with their discharge valves closed for planned maintenance. Even though adequate core cooling is met, the shift manager determined that the "A" & "C" RHR pumps need to be returned to service. Radiation protection has determined that the dose rate by the pump discharge valves is 60 Rem / hr. To meet the emergency exposure guideline the maximum time an individual has to open the discharge valves is __(a)__ minutes and his emergency exposure must be approved by the __(b)__ .

A. (a) 10
(b) Emergency Director

B. (a) 10
(b) TSC Manager

C. (a) 20
(b) Emergency Director

D. (a) 20
(b) TSC Manager

Proposed Answer: A (a) 10
(b) Emergency Director

Explanation (Optional): EAP-15, Emergency Radiation Exposure Criteria and Control limits the maximum dose to protect equipment to 10 Rem and this exposure must be approved by the emergency director.

Technical Reference(s): EAP-15, Emergency Radiation Exposure Criteria and Control

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.3.4 Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10) RO 2.5 / SRO 3.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X___

10 CFR Part 55 Content:	55.41	_____
	55.43	__4___

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295009 AA2.03	
Importance Rating	_2.9_	_2.9_

Proposed Question: 4 / 3

The reactor is critical at 160 psig. The feedwater level control system has been removed from service. Reactor water level has been stable and is being controlled via the "A" CRD pump and reactor water clean up (RWCU) blowdown at a blowdown rate of 60 gpm. Several minutes ago the "A" CRD pump tripped. There are no CRD accumulators alarms are present. The operator must __(1)__ RWCU blowdown rate to prevent __(2)__ .

A. (1) Reduce
(2) a HPCI / RCIC injection signal

B. (1) Reduce
(2) a reactor scram

C. (1) Raise
(2) a HPCI / RCIC trip signal

D. (1) Raise
(2) a reactor scram

Proposed Answer: B. (1) Reduce
(2) a reactor scram

Explanation (Optional):

A. This condition will never occur. The water level reduction will be terminated when level reaches 177 and RWCU isolates. This is above the 126.5 level for RCIC and HPCI start.

B. Correct - if the operator does not reduce the blowdown then a reactor scram / GP II / RWCU isolation will occur.

C. This condition will not occur. If the blowdown is raised the level will drop, not rise.

D. If the blowdown is increased the low level scram / isolations will occur.

Technical Reference(s):SDLP-12, RWCU, OP-23, OP-65, AOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: AA2. Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) AA2.03 Reactor water cleanup blowdown rate 2.9 / 2.9

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__5___
	55.43	______

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295010 AA1.01	
Importance Rating	_3.4_	_3.5_

Proposed Question: 5 / 4

The plant is operating at 100% power when an operator mistakenly isolates the reactor building closed loop cooling (RBCLC) to the "A" Drywell Cooler. The operator notifies the control room of the mistake and states that he can not reestablish any flow to the "A" Drywell Cooler. What effect does this have on the containment and what operator actions would have to be taken.

A. Drywell temperature and pressure will remain steady because the redundant "B" drywell cooler is in service. No operator action will be required.

B. Drywell temperature and pressure will rise. Operator action will be required to maintain drywell pressure below 2.7 psig.

C. Drywell temperature and pressure will remain steady due to the stored heat capacity in the cooling water of the "A" drywell cooler. No operator action is required.

D. Drywell temperature and pressure will rise. Operator action is required to manually start the fourth fan on the "B" drywell cooler which will then be able to maintain drywell temperature below 135°F.

Proposed Answer: B. Drywell temperature and pressure will rise. Operator action will be required to maintain drywell pressure below 2.7 psig.

Explanation (Optional):

A. A loss of 50% cooling in the drywell at 100% power will result in a rapid heat up and pressure rise (minutes).

B. A loss of 50% cooling in the drywell at 100% power will result in a rapid heat up and pressure rise (minutes). The operator will enter EOP-4 on high drywell temperature and be required to maintain drywell pressure below 2.7 psig.

C. There is not sufficient heat capacity to maintain drywell temperature less than 135 by starting the forth fan on the "B" drywell cooler.

D. Starting the forth fan on the "B" cooler will not be able to maintain temperature below 135.

Technical Reference(s): EOP-4, OP-53, "DRYWELL VENTILATION AND COOLING."

Proposed references to be provided to applicants during examination: None

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.7 / 45.6) AA1.01 Drywell ventilation/cooling 3.4 / 3.5

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New ___x___

Question History: Last NRC Exam ________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1_
	Group #	__1__	__1_
	K/A #	295014 AK2.11	
	Importance Rating	_3.6_	_3.7_

Proposed Question: R5

The plant is operating at 100% power when the "B" reactor recirculation pump trips. Which ONE of the following conditions describes when a manual reactor scram needs to be initiated?

A. Immediately if APRM peak-to-peak oscillations of greater than 10% occur.
B. Immediately if core plate differential pressure oscillations are greater than 2.9 psid.
C. Immediately if a LPRM upscale alarms occurs.
D. Following determination that the MCPR Safety Limit has been exceeded.

Proposed Answer: A. Immediately if APRM peak-to-peak oscillations of greater than 10% occur.

Explanation (Optional):
A. The criteria is greater than 10% APRM peak-to-peak oscillations.
B. Core plate oscillation is not a criteria for a manual scram.
C. LPRM periodic upscale and downscale alarms, not a single LPRM.
D. Safety limit violations do not require a scram.

Technical Reference(s): AOP-8 Loss of Coolant Flow, AOP-32Unexplained Reactivity Change

Proposed references to be provided to applicants during examination: None

Learning Objective: AK2. Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: (CFR: 41.7 / 45.8) AK2.11 Recirculation flow control 3.6 / 3.7

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __7___
55.43 __5___

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__1__	__1__
	K/A #	295014 AA1.01	
	Importance Rating	_4.0_	_4.1_

Proposed Question: 22 / 6

The plant was operating on a 102% rod line at 85% power and 60% drive flow. Several minutes ago a Hi-Hi level annunciator for the 6B feedwater heater alarmed. All average power range monitors (APRM) are currently at 103% power and all APRM rod block alarms are alarming. The only operator actions taken were to silence and acknowledge annunciators. Which statement correctly assesses the plant condition and gives the correct operator action?

A. A loss of feedwater heating has occurred and power must be rapidly reduced to 85% with reactor recirculation flow.

B. A loss of feedwater heating has occurred and control rods must be inserted to reduce reactor power to 85% power.

C. The APRMs did not initiate an automatic reactor scram, a manual reactor scram is required.

D. The APRMs indicate that core instabilities are present, a manual reactor scram is required.

Proposed Answer: C. The APRMs failed to generator an automatic reactor scram, a manual reactor scram is required. (EOP-2, entry condition)

Explanation (Optional):

A. AOP-62 requires power to be rapidly reduced to 20% below the initial power level. Reduce power to 65%.

B. AOP-62 requires rod insertion until below the 100% rod line. Inserting control rods back to 85% power will only reduce the rod line back to 101%.

C. EOP-2 requires that if reactor has not scrammed then a manual scram is required. The APRM scram set point is 0.58*Wd + 66. This is (.58(62))+66= 101

D. The APRMs are not exhibiting indication of instability. ARPMs are stable not experiencing 10% peak-to-peak oscillations.

Technical Reference(s):AOP-62, EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.7 / 45.6) AA1.01 RPS 4.0/4.1

SDLP-07C, Power Range Monitors
1.07, TS setpoints for (c) APRMs

Question Source:

Bank # _______
Modified Bank # _______ (Note changes or attach parent)
New ___X____

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41 __7___

55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295015 AK1.01	
Importance Rating	_3.6_	_3.9_

Proposed Question: 8 / 7

The plant has just experienced a scram. All control rods fully inserted with the exception of control rod 26-27, which inserted to position 24. Subsequent attempts to insert the control rod were not successful. The control room supervisor orders a cool down in accordance with the EOPs. What actions are required, prior to the cool down to ensure that the reactor will remain shutdown during the cool down.

A. Control rod 26-27 must be fully inserted prior to beginning the cool down.
B. Standby liquid control must be injected prior to beginning the cool down.
C. Reactor engineering must perform calculations to prove the reactor will remain shutdown prior to beginning the cool down.
D. No actions are necessary, the reactor will remain shutdown during the cool down.

Proposed Answer: D. No actions are necessary, the reactor will remain shutdown during the cool down.

Explanation (Optional): shutdown margin requires that the reactor will remain shutdown under all conditions without boron injection with one control rod fully withdrawn (or any other position), provided all other control rods are inserted to or beyond position 02.

Technical Reference(s): EP-6, Backup control rod insertion

Proposed references to be provided to applicants during examination: None

Learning Objective: AK1. Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: (CFR: 41.8 to 41.10)
AK1.01 Shutdown margin 3.6* / 3.9*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __x___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __8___
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__1__	__1__
	K/A #	295024 EA2.01	
	Importance Rating	_4.2_	_4.4_

Proposed Question: 12 / 8

A major transient has just occurred, no operator actions have been taken. The following plant conditions exist:

Torus pressure ....28 psig
Torus water level ....16 feet
Torus water temperature ....155 °F
DW Pressure ....30 psig
DW Temperature ....290°F
RPV water level ....100 inches
Reactor pressure ....500 psig and dropping

Considering the above conditions, _(1)_ to prevent the _(2)_ from being exceeded.

A. (1) Start Torus venting through SBGT
(2) Pressure Suppression Pressure Limit

B. (1) Start drywell venting through SBGT
(2) Primary Containment Pressure Limit

C. (1) Start Drywell sprays
(2) Drywell Design Temperature

D. (1) Start Torus sprays
(2) SRV Tail Pipe Limit

Proposed Answer: C. (1) Start Drywell sprays
(2) Drywell Design Temperature

Explanation (Optional): A & B These methods should be used during normal operation to maintain the containment pressure below the drywell pressure set point or post accident when conditions in EOP-4 are met. Drywell / Torus pressure conditions are not met to perform post accident containment venting.

C. DW Sprays must be initiated before the drywell design temperature is reached.

D. Initiating torus spray will only move the plant closer to the limit.

Technical Reference(s): EOP-4, Primary Containment Control; EP-6, Post Accident Containment Venting.

Proposed references to be provided to applicants during examination: EOP-4

Learning Objective:	EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.10 / 43.5 / 45.13) EA2.01 Drywell pressure 4.2*/ 4.4*

MIT-301.11E, EOP-4

LO 1.03,	Identify situations where it is appropriate to enter other procedure concurrently - Task 344169, Spray DW
LO 1.05	Explain basis for any step in the EOP - Task 344169, Spray DW
LO 1.07	Explain the basis and demonstrate the use of all figures associated with EOP-4, Task 344132, Monitor and control DW temperature.

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41 __10__
55.43 __5___

Comments:

The learning objectives do not distinguish between SRO and RO for EOPs. The Objectives state "The operator should be able to ... "

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295025 K1.04	
Importance Rating	_3.6_	_3.9_

Proposed Question: 25 / 9

The plant has just scrammed after an extended full power run. The feedwater pumps are maintaining RPV level and RPV pressure is 970 psig using the bypass valves. NCO2 has just reported a trip of the "A"&"B" circulating water pumps and has taken appropriate action for the loss of these pumps in accordance with AOP-1, "Scram." If no additional operator action is taken how will RPV pressure respond?

A. will drop from 970 psig as decay heat drops

B. will remain at or below approximately 970 psig

C. will rise to 1135 psig

D. will rise to 1145 psig

Proposed Answer: C. will rise to 1135 psig

Explanation (Optional): The "C" Circ water pump was lost on the scram. When the A&B circ water pump is lost AOP-1 directs closing the MSIVs. If the MSIVs are closed then RPV pressure will be controlled at the lowest relief valve setting 1135psig. Since one relief valve will pass about 8% total steam flow only one will be open and the pressure will remain at that setting until it decays away at a later time.

Technical Reference(s):SDLP 02J pp 23 and SDLP-94C figure 8

Proposed references to be provided to applicants during examination: None

Learning Objective: EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: (CFR: 41.8 to 41.10) EK1.04 Decay heat generation 3.6 / 3.9

Question Source: Bank # FitzPatrick Requalification 753

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __8__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1___
Group #	__1__	__1___
K/A #	295031 EA1.06	
Importance Rating	_4.4_	__4.4__

Proposed Question: 15 / 10

A small break LOCA has occurred and no operator action has been taken. HPCI is out of service, RCIC is injecting into the reactor vessel and all low pressure ECCS pumps have started from high drywell pressure except the "B" core spray pump which failed to automatically start. One hundred (100) seconds ago the reactor water level was 177 inches, current reactor water level is 75 inches and dropping. Which statement correctly describes the operation of the Automatic Depressurerization System (ADS)?

A. All 7 ADS valves will open in 34 seconds.
B. All 7 ADS valves will open 134 seconds after reactor water level drops to 59.5 inches.
C. All 7 ADS valves will open 134 seconds after the "B" core spray pump is manually started.
D. All 7 ADS valves will open immediately after the "B" core spray pump is manually started.

Proposed Answer: B. All 7 ADS valves will open in 134 seconds after reactor water level decreases to 59.5 inches.

Explanation (Optional):

The following condition are required for ADS to automatically open all 7 valves. Any low pressure ECCS pump running, reactor low 177 and low low low level 59.5 after 134 second timer times out.

Technical Reference(s): SDLP-02J, ADS

Proposed references to be provided to applicants during examination: NONE

Learning Objective: EA1. Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.7 / 45.6) EA1.06 Automatic depressurization system 4.4* / 4.4*

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7___
55.43 _____

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295031 EK2.04	
Importance Rating	_4.0_	_4.1_

Proposed Question: 17 / 11

A spurious main steam isolation has occurred, all MSIV's have closed. All control rods have inserted and RPV pressure is being controlled 800 to 1000 psig using safety / relief valves. RPV level has just reached 126.5 inches. Which one of the following actions would you expect to occur at this level.

A. A & B Reactor Recirculation Pump Trips.
B. Core Spray System auto start.
C. Reactor Core Isolation Cooling auto start.
D. Standby Gas Treatment System auto start.

Proposed Answer: C. Reactor Core Isolation Cooling auto start.

Explanation (Optional):
A. RR pump will trip at RPV level of 105.4 inches or may already be tripped due to the high pressure from the initial MSIV closure.
B. Core Spray starts at 59.5 inches.
D. SBGT starts at 177 inches.

Technical Reference(s):SDLP-02B, RR and SDLP-02H, RPV Level Instrumentation.

Proposed references to be provided to applicants during examination: None

Learning Objective: EK2. Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: (CFR: 41.7 / 45.8) EK2.04 Reactor core isolation cooling: Plant-Specific. 4.0 / 4.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7___
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295037 EK1.03	
Importance Rating	_4.2_	_4.4_

Proposed Question: 18 / 12

The reactor failed to scram and the following conditions exist:

RPV water level..........................195 inches
Reactor Power............................18%
Torus Level................................16 ft
Torus water temperature...........115 °F
RPV Pressure ....1000 psig
Safety / Relief Valves................Cycling to control Reactor pressure.
Reactor Recirculation pumps....Tripped
All emergency and normal sources of reactor makeup water are available at full capacity.

In accordance with EOP-3 standby liquid control (SLC) will be initiated based on exceeding the __(1)__. SLC tank level must drop by _(2)_ to ensure that the reactor will stay shut down under all conditions.

A. (1) Heat Capacity Temperature Limit
(2) 22% SLC tank level

B. (1) Heat Capacity Temperature Limit
(2) 46% SLC tank level

C. (1) Boron Injection Initiation Temperature
(2) 22% SLC tank level

D. (1) Boron Injection Initiation Temperature
(2) 46% SLC tank level

Proposed Answer: D. (1) Boron Injection Initiation Temperature
(2) 46% SLC tank level

Explanation (Optional): The Boron Injection Initiation Temperature has been exceeded. At 18% power this correlates to a torus water temperature of 110 °F. 22% SLC tank level will not ensure that the reactor will stay shutdown under all conditions. 46% ensures that a hot / 100% Xenon reactor will be shutdown. 46% tank level will account for the effects of cool down and decay of Xenon.

The heat capacity temperature limit has not been exceeded. At a torus level of 16 feet, RPV pressure of 1000 psig the HCTL is 180 °F Torus water temperature.

Technical Reference(s): EOP-3, "Failure to scram," MIT 301.11d, "Failure to Scram"

Proposed references to be provided to applicants during examination: EOP-3, "Failure to scram

Learning Objective: EK1. Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.8 to 41.10) EK1.03 Boron effects on reactor power (SBLC) 4.2 / 4.4*

MIT 301.11d, EOP-3, "Failure to scram

1.06, Explain the basis of and demonstrate the use of all figures associated with EOP-3

Task 344058, Monitor and control reactor power

Task 344228, Monitor and control torus temperature.

2.04 Explain the reason or purpose for any step in Emergency RPV Depressurization

Question Source: Bank # ________

Modified Bank # ________ (Note changes or attach parent)

New ___X____

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _8-10_

55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	50000 EK3.07	
Importance Rating	_3.1_	_3.7_

Proposed Question: 20 / 13

The unit has experienced a large break loss of coolant accident which depressurized the reactor. The following conditions exist.

Low Pressure ECCS................................All pumps are running
Drywell Hydrogen concentration 7%
Drywell Oxygen concentration 7%
Torus Hydrogen concentration 5%
Torus Oxygen concentration...................3%
Offsite release rate ....Will exceed the general emergency release rate

The control room supervisor orders that the drywell be vented and purged per EP-6, through the torus. Based on the Hydrogen & Oxygen concentrations is the action correct and why or why not?

A. No, adequate core cooling is assured, no venting is required until torus Hydrogen is greater than or equal to 6%.
B. No, the drywell can not be vented if the release rate exceeds the general emergency releases rate.
C. Yes, the drywell must be vented to prevent a deflagration and venting through the torus will minimize the radioactive release.
D. Yes, venting the drywell will allow cooler nitrogen to purge the drywell thus slowing down the Zirconium-water reaction which will reduce the hydrogen in the drywell.

Proposed Answer: C. Yes, the drywell must be vented to prevent a deflagration and venting through the torus will minimize the radioactive release.

Explanation (Optional): A & B Based on the Hydrogen and Oxygen concentration the drywell must be vent to prevent a deflagration.
D. Purging the drywell will not have a significant effect on the Zirconium-water reaction.

Technical Reference(s): EOP-4, 4a, EP-6, and MIT-301.11e

Proposed references to be provided to applicants during examination: EOP-4, 4a

Learning Objective: EK3. Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: (CFR: 41.5 / 45.6) EK3.07 Operation of drywell vent 3.1/ 3.7

MIT-301.11E, EOP-4, Primary Containment Control
1.03, Identify situations where it is appropriate to enter other procedures concurrently.
1.05, Explain the basis or purpose for any step in EOP-4.

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__5__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	_____	__1__
Group #	_____	__2__
K/A #	2950001 AK1.01	
Importance Rating	_3.5_	_3.6_

Proposed Question: 27 / 14

The plant is operating at 100 % when I&C inadvertently trips both reactor recirculation pumps while working on the ARI/RPT logic. Where will the reactor be on the Power-Flow Map and what actions are required.

A. The reactor will be in the Power-flow Map BUFFER ZONE. Monitor nuclear instrumentation, for indications of thermal-hydraulic instability.
B. The reactor will be in the Power-flow Map BUFFER ZONE. Manually scram the reactor.
C. The reactor will be in the Power-flow Map EXCLUSION ZONE. Manually insert control rods to exit the exclusion zone.
D. The reactor will be in the Power-flow Map EXCLUSION ZONE. Manually scram the reactor.

Proposed Answer: D. The reactor will be in the Power-flow Map EXCLUSION ZONE. Manually scram the reactor

Explanation (Optional): D. The trip of both RWR pumps will result in the plant being at about 50% power on the natural circulation line of the power to flow map. AOP-8 requires a manual scram when both RWR pumps are tripped.

A. The reactor not be in the buffer zone.
B. The reactor will not be in the buffer zone.
C. The actions listed in C are correct if both RR pumps have not tripped. However, the first step in AOP-8 requires a manual scram if both RWR pumps trip.

Technical Reference(s):TS Basis 3.5.J, AOP-8

Proposed references to be provided to applicants during examination: None

Learning Objective: AK1. Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.8 to 41.10) AK1.01 Natural circulation 3.5 / 3.6

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:	55.41	__8__
	55.43	__5__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	295002 AK2.07	
Importance Rating	_3.1_	_3.1_

Proposed Question: 28 / 15

The plant is operating at 100 % power when annunciator 09-6-1-29 CNDSR VAC LO alarms. Condenser vacuum has slowly dropped to 25 inches of Hg and generator output has dropped by 3 MWe. A NLO identified a small tear in the expansion boot between the condenser and LP turbine hood. How has the offgas flow changed (prior to any operator action) and what actions, in addition to a power reduction, will the operator take in response to this event.

A. Offgas flow has dropped, trip hydrogen addition and start the condenser air removal pumps.
B. Offgas flow has dropped, trip the turbine, scram the reactor and close the main steam line isolation valves.
C. Offgas flow has risen, trip hydrogen addition and start the condenser air removal pumps.
D. Offgas flow has risen, place the spare steam jet air ejectors in service.

Proposed Answer: D. Off gas flow has INCREASED, place the spare steam jet air ejector in service.

Explanation (Optional): A. & B. The off gas flow will increase not decrease with a condenser boot tear. There is more non Condensible gas entering the condenser and the vacuum decreases.

C. Condenser air removal pumps discharge to the 1.75 minute holdup pipe, which is not designed for explosion pressure. For this reason, operation of condenser air removal pumps is not permitted if reactor power is greater than 5%. Power is greater than 5%.

Technical Reference(s): AOP-31, loss of condenser vacuum

Proposed references to be provided to applicants during examination: None

Learning Objective: AK2. Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: (CFR: 41.7 / 45.8) AK2.07 Offgas system 3.1 / 3.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X___

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__2__	__1__
	K/A #	295003 AA1.03	
	Importance Rating	_4.4_	_4.4_

Proposed Question: 1 / 16

A station blackout has occurred. What action is necessary to ensure the plant can be safely shutdown.

A. The high pressure coolant injection (HPCI) system should be used preferentially over the reactor core isolation cooling (RCIC) system for RPV make up because of the higher flow rate.

B. The RCIC system should be used preferentially over HPCI for RPV makeup because RCIC is less likely to cycle on and off due to RPV water level.

C. Align the RCIC suction to the Torus since there is no power available for make up water to the CST.

D. Verify that all fire doors in the HPCI and RCIC room are closed so that a steam line break will not result in the loss redundant systems simultaneously.

Proposed Answer: B. The reactor core isolation cooling (RCIC) system should be used preferentially over HPCI for RPV makeup because RCIC is less likely to cycle on and off due to RPV water level.

Explanation (Optional):
A. HPCI has a larger capacity and will cycle more than RCIC. This will result in less battery operational time.
C. RCIC low CST level should be bypassed because operation with torus water at elevated temperatures will reduce the RCIC reliability.
D. The doors are required to be opened to reduce local area temperatures because of Steam leakage from HPCI and RCIC turbine shaft seals that could be encountered. The leakage could be caused by a lack of gland seal suction or higher than normal turbine exhaust pressures.

Technical Reference(s):AOP-49, Station Blackout

Proposed references to be provided to applicants during examination: None

Learning Objective: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.7 / 45.6) AA1.03 Systems necessary to assure safe plant shutdown 4.4*/ 4.4

Question Source:
Bank # ________
Modified Bank # _____ (Note changes or attach parent)
New __X__

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	295004 Ak3.02	
Importance Rating	_2.9_	_3.3_

Proposed Question: 29 / 17

The plant is operating at 100% with operators performing AOP-22, "DC Power System A Ground Isolation," due to a ground on the "A" station battery. The next breaker to be opened is the supply for the 10700 bus breaker Control Power. When this breaker is OPENED, which one of the following statements correctly describe the effects or actions that this will have on the 10700 bus?

A. All 10700 bus breaker protection trips will operate normally because the bus logic power has automatically swapped to "B" 125 VDC.

B. All 10700 bus breaker red / green position indicating lights will still indicate breaker positions because the logic power has automatically swapped to "B" 125 VDC.

C. All 10700 bus breakers will open if originally closed due to a loss of "A" 125 VDC control power.

D. All 10700 bus breakers will lose red / green position indicating lights because the breakers have lost "A" 125 VDC control power.

Proposed Answer: D. All 10700 bus breakers will lose red / green position indicating lights because the breakers have lost "A" 125 VDC control power.

Explanation (Optional): A. & B. There is no automatic swap of 125 VDC control power on bus 10700, and the red and green lights will be lost when 125 VDC is lost.

C. The breakers will not automatically open on loss of DC control power.

Technical Reference(s): SDLP-71B, AOP-22, OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: AK3. Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.5 / 45.6) AK3.02 Ground isolation/fault determination 2.9 / 3.3

Question Source: Bank # FitzPatrick Requalification bank 0869

The plant is operating at 100% with operators performing AOP-22, DC POWER SYSTEM A GROUND ISOLATION due to a ground on the "A" station battery. The next breaker to be opened is the supply for 10700 BKR Control Power (71DCA3 Crkt 24). When this breaker is OPENED, Which one of the following statements will occur?

a) All 10700 bus breaker protection trips will operate normally.

b) All 10700 bus breakers will open if originally closed.

c) All 10700 bus breakers can be tripped locally if closed.

d) All 10700 bus breaker position indication lights (red and green) will still indicate breaker positions.

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	295008 AA1.01	
Importance Rating	_3.7_	_3.7_

Proposed Question: 30 / 18

The reactor protection system initiated a scram, all blue scram lights are lit on the full core display. However, little rod motion occurred, power reduced to 95% and the plant remains stable with the following indications:

Reactor Power............................95%
"A" RWR Pump Speed...............90%
"B" RWR Pump Speed...............91%
Mode Switch...............................SHUTDOWN
Alternate Rod Insertion...............INITIATED
Turbine.......................................ON LINE
Reactor Pressure........................1035 psig
Reactor Level.............................200 inches
Drywell Temperature..................130 °F
Drywell Pressure........................2.2 psig
Torus Water Temperature..........75 °F
Torus Pressure ....0.3 psid

What operator actions must be taken and why?

A. Inject standby liquid control because the boron injection temperature has been exceeded.
B. Immediately trip both RWR pumps to achieve a rapid power reduction.
C. Reduce recirculation flow to minimum then trip both RWR pumps to prevent the turbine from tripping.
D. Vent the scram air header to open the scram inlet and outlet valves.

Proposed Answer: C. Reduce recirculation flow to minimum then trip both RWR pumps to prevent the turbine from tripping.

Explanation (Optional):
A. The Boron injection temperature has not been exceeded.
B. In this case the RWR should be run to minimum speed to prevent a RPV high level transient from tripping the turbine and forcing all the heat load into containment.
D. All the scram inlet and outlet valves are open as indicated by the blue lights on the full core display.

Technical Reference(s): EOP-3, MIT 301.11d, page 5

Proposed references to be provided to applicants during examination: EOP-3

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL: (CFR: 41.7 / 45.6) AA1.01 Reactor water level control: Plant-Specific 3.7 / 3.7

MIT 301.11d
1.07 Explain the basis or purpose for any step in EOP-3

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__1__
Group #	__2__	__1__
K/A #	295013 AK3.02	
Importance Rating	_3.6_	_3.8_

Proposed Question: R19

Which one of the following is the purpose of the Heat Capacity Temperature Limit (HCTL) Curve?

A. To prevent exceeding the Primary Containment Pressure limit during a DESIGN BASIS LOCA before the blowdown energy transfer is within the capacity of the containment vent.

B. To prevent exceeding the Primary Containment Pressure limit during EMERGENCY DEPRESSURIZATION before the blowdown energy transfer is within the capacity of the containment vent.

C. To prevent dynamic pressure loads from exceeding the structural limits of the torus and submerged suppression chamber components during an EMERGENCY DEPRESSURIZATION.

D. To prevent dynamic pressure loads from exceeding the structural limits of the torus and submerged suppression chamber components during a DESIGN BASIS LOCA.

ANSWER: B. To prevent exceeding the Primary Containment Pressure limit during EMERGENCY DEPRESSURIZATION before the blowdown energy transfer is within the capacity of the containment vent.

Explanation (Optional): A. Incorrect because the HCTL is based on an EMERGENCY DEPRESSURIZATION not a DESIGN BASIS LOCA. The analysis for the HCTL is based on a heat balance 2 minutes after shutdown. The LOCA loads could occur simultaneously with the head addition and should be protected by TS by limiting the torus water temperature and limiting operation at high water temperatures. (120F)

Technical Reference(s): MIT-301.11B pp. 11

Learning Objective: AK3. Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.5 / 45.6) AK3.02 Limiting heat additions 3.6 / 3.8

Question Source: Bank # INPO 9503

Which one of the following is the purpose of the Heat Capacity Temperature Limit (HCTL) Curve?

a. To prevent exceeding the Primary Containment Pressure limit during EMERGENCY DEPRESSURIZATION before the blowdown energy transfer is within the capacity of the containment vent.

b. To prevent exceeding the Primary Containment Pressure limit during a DESIGN BASIS LOCA before the blowdown energy transfer is within the capacity of the containment vent.

c. To prevent dynamic pressure loads from exceeding the structural limits of the suppression pool and submerged suppression chamber components during an EMERGENCY DEPRESSURIZATION.

d. To prevent dynamic pressure loads from exceeding the structural limits of the suppression pool and submerged suppression chamber components during a DESIGN BASIS LOCA.

Answer a

Reference: ..295013.K3.02

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _x_
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 __5___
55.43 ______

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295013 AA2.02	
Importance Rating	_3.2_	_3.5_

Proposed Question: 21 / 20

The NCO2 has been directed to maintain RPV pressure between 800 and 1000 psig using safety relief valves (SRVs). You observe the NCO2 continuously opening and closing the "A" SRV to maintain pressure. Is this an acceptable method of cycling the SRV and what is the basis for this operation?

A. NO. The operator should cycle through the SRVs in the following order A, J, K, G, E, D, C, F, H, L, B to prevent high local pool temperatures that could result in inefficient pool cooling.

B. NO. The operator should cycle through the SRVs in the following order A, J, K, G, E, D, C, F, H, L, B to equally deplete all Nitrogen accumulators.

C. NO. The operator should cycle through the SRVs in the following order A, B, C, D, E, F, G, H, J, K, L to prevent high cyclic fatigue loads, due to chugging, on an individual valve.

D. YES. Cycling only "A" SRV will limit the number of SRV actuations on the other SRVs and minimize the cost required to replace the valve(s) during the next refueling outage.

Proposed Answer: A. NO. The operator should cycle through the SRVs in the following order A, J, K, G, E, D, C, F, H, L, B to prevent high local pool temperatures that could result in inefficient pool cooling.

Explanation (Optional):
B. There is no such requirement.
C. Chugging should not occur if the EOPs are followed.
D. Cycling only one SRV could result in high local pool temperatures that could result in inefficient pool cooling.

Technical Reference(s): MIT 301.11C

Proposed references to be provided to applicants during examination: EOP-2, "RPV Control"

Learning Objective: AA2. Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) AA2.02 Localized heating/stratification 3.2 / 3.5

MIT 301.11C LO EO-1.06

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__10_
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1_
Group #	__2__	__1_
K/A #	295016 AK3.03	
Importance Rating	_3.5_	_3.7_

Proposed Question: 23 / 21

The shift manager has just ordered a control room evacuation in accordance with AOP-43, "Plant Shutdown From Outside the Control Room." When you arrive at the auxiliary shutdown panel 25 RSP you place the isolation switch for LPCI INBOARD INJECTION VALVE (10MOV-25B) in LOCAL. Reactor water level is 50 inches and reactor pressure is 700 psig. Reactor water level and reactor pressure are dropping. With the isolation switch in LOCAL which statement below describes the operation of the LPCI INBOARD INJECTION VALVE (10MOV-25B) valve as reactor pressure and reactor water level drop.

A. Will automatically open when reactor pressure is less than 450 psig.
B. Will automatically open when the "B" residual heat removal (RHR) pump is started.
C. Must be manually opened from the auxiliary shutdown panel.
D. Must be manually opened at the valve.

Proposed Answer: C. Must be manually opened from the auxiliary shutdown panel.

Explanation (Optional): When the isolation switch for the 10MOV-25B is placed in LOCAL this disables all interlocks and will only allow operation of this valve from the auxiliary shutdown panel. In addition, the valve will not open when the pump is started, the interlock is bases on an ECCS signal and pressure.

Technical Reference(s): AOP-43, SDLP-10 RHR

Proposed references to be provided to applicants during examination: None

Learning Objective: AK2. Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: (CFR: 41.5 / 45.6) AK3.03 Disabling control room controls: 3.5 / 3.7

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295017 AK3.02	
Importance Rating	_3.3_	_3.2_

Proposed Question: 10 / 22

A loss of coolant accident has occurred. The following conditions are present:

Reactor Building Ventilation..ISOLATED
Reactor Building Ventilation Exhaust Radiation................1 x 10E5 cpm
"A" Standby Gas Train..OPERATING
Reactor Building to Atmosphere Differential Pressure..... negative (-)1.1 inches water
Turbine Building Ventilation ....ISOLATED
Turbine Building Exhaust Radiation.................................3 x 10E4 cpm
Offsite Release ....Above the ALERT Level

Which ventilation system should be reestablished and why?

A. Reestablish the turbine building ventilation to prevent an unmonitored ground level release of radioactivity.
B. Reestablish the turbine building ventilation to filter the turbine building exhaust.
C. Reestablish the reactor building ventilation to prevent an unmonitored ground level release of radioactivity.
D. Reestablish the reactor building ventilation to reduce the reactor building area and equipment temperatures.

Proposed Answer: A. Reestablish the Turbine building ventilation to prevent an unmonitored ground level release of radioactivity.

Explanation (Optional): The turbine building is not a leak tight building. Restarting the turbine building vent will result in an elevated release point for any radioactivity in the building. In addition, EOP-4 states that if reactor building exhaust radiation is greater than 1E4 then isolate reactor building vents.

Technical Reference(s): GE EOP manual page 6-2. There was no basis document for EOP-6. FitzPatrick must very this answer.

Proposed references to be provided to applicants during examination: None

Learning Objective: AK3. Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.5 / 45.6) AK3.02 Plant ventilation 3.3 / 3.5

Question Source: Bank # INPO 6578

Which of the following explains why DEOP 300-2, Radioactive Release Control, directs the operator to restart the Turbine Building Ventilation, if it is shutdown?
to prevent an unmonitored ground level release of radioactivity.
to maintain a positive pressure inside the turbine building.
to reduce the turbine building area and equipment temperatures.

to filter the air in the turbine building before release to the environment.
Reference: ..295017.K3.02

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	295018 A2.01	
Importance Rating	_3.3_	_3.4_

Proposed Question: 31/ 23

The "B" & "D" emergency diesel generators (EDGs) are tagged out of service. A loss of coolant accident and a loss of offsite power have just occurred. "A" & "C" EDGs are operating and required for core cooling. The ESW LOW FLOW annunciator is alarming due to a failure of the "A" emergency service water (ESW) pump. The "A" & "C" EDGs have a jacket water temperature of 190°F and rising. How will the "A" & "C" EDG respond to this condition and what operator actions, if any, are required? The "A" & "C" EDGs...

A. will automatically trip when the jacket water temperature reaches 205°F, attempt to start the emergency service water (ESW) pump.
B. must be manually tripped before the jacket water temperature reaches 205°F, have the control room shutdown the "A" & "C" EDGs.
C. will continue to run at jacket water temperatures above 205°F, align the fire protection system to supply the "A" & "C" EDGs.
D. high jacket water temperature trip must be bypassed before reaching 205°F, install jumpers to bypass the high temperature trips on the "A" & "C" EDGs.

Proposed Answer: C. will continue to run at jacket water temperatures above 205°F, align the fire protection system to supply the "A" & "C" EDGs.

Explanation (Optional):
A. If a LOCA signal is present the EDG will not trip on high jacket water temperature.
B. The EDG should not be shutdown if they are required for core cooling. In addition, at higher temperature the EDG will continue to run but at reduced load. The EDGs can not be shutdown from the control room.
C. Correct - ARP 93ECP-A-12 and OP-22
D. The trip is automatically bypassed under LOCA conditions.

Technical Reference(s): ARP 93ECP-A-12, OP-22

Proposed references to be provided to applicants during examination: None

Learning Objective: AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.10 / 43.5 / 45.13) AA2.01 Component temperatures 3.3 / 3.4

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 _10__	
	55.43 _____	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2___	__2__
K/A #	295020 AA2.01	
Importance Rating	_3.6_	_3.7_

Proposed Question: 33 / 24

The Unit is operating at 100% power when a spurious loss of RPS bus "A" occurs. The following conditions are present after the loss of the "A" RPS bus.

Drywell Temperature ....130°F
Drywell Pressure...2.1 psig
Torus Water Temperature ....72°F
Torus Pressure ....0.8 psig
Torus Level ....13.98 Feet
Drywell and Torus Oxygen Concentration ....2.1 volume percent

Based on these conditions what operator actions are required after the "A" RPS bus is restored per AOP-59, "Loss of RPS Bus A Power."

A. Reopen the reactor building closed loop cooling Drywell Cooler "A" Inlet and outlet valves to ensure that the drywell pressure remains below 2.7 psig.
B. Vent the drywell through standby gas to ensure that the drywell pressure remains below 2.7 psig.
C. Vent the torus to establish drywell to torus differential pressure within the Technical Specification required value.
D. Start drywell makeup using CAD Train A to maintain the oxygen concentration within the Technical Specification required value.

Proposed Answer: C. Vent the Torus to establish drywell to torus differential pressure within the Technical Specification required value.

Explanation (Optional):

A. The Drywell Cooler valves do not go closed on a loss of RPS.
B. By venting the Drywell through SBGT the drywell to torus dP will be reduced further.
C. Differential pressure is 1.3, TS requires > 1.7 psid.
D. The Oxygen concentration is allowable by TS (less than 4.0 volume Percent)

Technical Reference(s): TS 3.7, OP-37, Containment Atmosphere Dilution System, AOP-59 Loss of RPS bus A Power

Proposed references to be provided to applicants during examination: None

Learning Objective: AA2. Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.10 / 43.5 / 45.13)
AA2.01 Drywell/containment pressure 3.6 / 3.7

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

Question History: Last NRC Exam ______
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 ____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	295022 Ak3.02	
Importance Rating	_2.9_	_3.1_

Proposed Question: 35 / 25

Five minutes ago both CRD pumps tripped. This condition will result in a loss of __(1)__ and may result in __(2)__ .

A. (1) Cooling water flow to the CRD
(2) degradation of the Graphitar seals

B. (1) Charging water flow to the CRD
(2) loss of scram capability

C. (1) Drive water flow to the CRD
(2) failure of the chromel / alumel temperature sensor in the position indication probe

D. (1) Exhaust water flow from the CRD
(2) failure of the collet housing from inter-granular stress corrosion cracking

Proposed Answer: A. (1) Cooling water flow to the CRD
(2) degradation of the Graphitar seals

Explanation (Optional): At elevated temperatures the Graphitar seals become brittle and can result in breakdown and increased scram times.

Technical Reference(s): SDLP-03A, AOP-69, Control Rod Drive Trouble

Proposed references to be provided to applicants during examination: None

Learning Objective: AK3. Knowledge of the reasons for the following responses as they apply to LOSS OF CRD PUMPS: (CFR: 41.5 / 45.6) AK3.02 CRDM high temperature 2.9 / 3.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __5___
55.43 ______

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295026 EA2.01	
Importance Rating	_4.1_	_4.2_

Proposed Question: 14 / 26

An ATWS has occurred. All control rods have been inserted using procedure EP-3, "Backup Control Rod Insertion." The following plant conditions exist:

Torus pressure ....5 psig
Torus water level ....14 feet
Torus water temperature ....180 °F
DW Pressure ....5 psig
DW Temperature ....145 °F
RPV water level ....100 inches
Reactor pressure ....1000 psig

Considering the above conditions, you must _(1)_ because the _(2)_ has been exceeded.

A. (1) Perform an Emergency Depressurization
(2) SRV Tail Pipe Limit

B. (1) Perform an Emergency Depressurization
(2) Heat Capacity Temperature Limit

C. (1) Perform RPV Flooding
(2) RPV Saturation Temperature Curve

D. (1) Start Standby Liquid Control
(2) Boron Injection Initiation Temperature

Proposed Answer: B. (1) Perform an Emergency Depressurization
(2) Heat Capacity Temperature Limit

Explanation (Optional):
A The SRV tail pipe limit has not been exceeded.
C. The RPV saturation temperature has not been exceeded.
D. All rods have been inserted reactor power is 0. There is no need to start standby liquid control.

Technical Reference(s): EOP-4, Primary Containment Control

Proposed references to be provided to applicants during examination: EOP-4, EOP-3, Failure to Scram

Learning Objective: EA2. Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) EA2.01 Suppression pool water temperature 4.1* / 4.2*

MIT-301.11E, EOP-4

LO 1.03,	Identify situations where it is appropriate to enter other procedure concurrently
LO 1.05	Explain basis for any step in the EOP
LO 1.07	Explain the basis and demonstrate the use of all figures associated with EOP-4

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41 __10__
55.43 __5___

Comments:

Examination Outline Cross-reference: Level

Level	RO	SRO
Tier #	1	1
Group #	2	2
K/A #	295028 EK2.02	
Importance Rating	3.2	3.3

Proposed Question: 36 / 27

A LOCA has just occurred and all plant systems functioned as designed. Based on the following list of plant parameters what reactor water level indication may be used by the operator.

Drywell Pressure ....35 psig
Drywell Instrument Run Temperatures ..320° F
Torus Pressure ....33 psig
Torus Water Temperature ....150° F
Reactor Pressure ....100 psig

A. Refuel Zone Level Indicating 200 inches
B. Narrow Range Level Indicating 164.5 inches
C. Fuel Zone Level Indicating negative (-)100 inches
D. RPV water level can not be determined

Proposed Answer: C. Fuel Zone Level Indicating -100 inches

Explanation (Optional):
A. Level is NOT above its minimum usable indication level.
B. Level is NOT above its minimum usable indication level.
C. Correct
D. The Fuel Zone Level indication is on scale and useable based on instrument run temperatures.

Technical Reference(s): EOP-2, "RPV Control"

Proposed references to be provided to applicants during examination: EOP-2, "RPV Control"

Learning Objective: EK2. Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: (CFR: 41.7 / 45.8) EK2.02 Components internal to the drywell 3.2 / 3.3

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

Question History: Last NRC Exam ______
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __7__
55.43 ____

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	295029 EK1.01	
Importance Rating	_3.4_	_3.7_

Proposed Question: 37 / 28

The control room supervisor has determined that torus water level and RPV pressure can not be maintained below the SRV Tail Pipe Level Limit. Why is an emergency depressurization required?

A. The SRV vacuum breakers will be submerged and will not limit the dynamic forces created by steam condensation in the SRVs tail pipes.
B. The high torus level will flood the HPCI and RCIC turbine exhaust lines and resulting in a common mode failure of emergency high pressure injection.
C. SRV operation could result in failure of the SRV tail pipes and direct containment pressurization.
D. Containment overpressurization could occur from the torus to drywell vacuum breakers being submerged.

Proposed Answer: C. SRV operation could result in failure of the SRV tail pipes and direct containment pressurization.

Explanation (Optional):
A The SRV vacuum breakers are located in the drywell and will not be submerged.
B. There are check valves in the HPCI and RCIC turbine exhaust lines that would prevent flooding.
D. Containment over pressure would not occur. Most of the non condensible is the nitrogen and to displace the nitrogen you would need a LOCA not a blowdown. In addition the containment vent would be able to keep up with the increased pressure during a blowdown.

Technical Reference(s): EOP-4, Primary Containment Control; EP-6, Post Accident Containment Venting.

Proposed references to be provided to applicants during examination: EOP-4

Learning Objective: EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL: (CFR: 41.8 to 41.10) EK1.01 Containment integrity 3.4 / 3.7

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:		55.41	__8__
		55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295030 EK3.03	
Importance Rating	_3.6_	_3.7_

Proposed Question: 16 / 29

An event has occurred which resulted in torus water level rapidly dropping to 5 feet. Torus level is being maintained at 5 feet. A manual reactor scram has been initiated and an emergency depressurization is in progress using group 2 pressure control systems. Under these plant conditions with torus level at 5 feet, which of the following statements is a valid RCIC operational concern.

A. The RCIC pump suction logic that transfers the RCIC suction to the CST on low Torus level must be defeated.
B. The RCIC system can not be operated in the RPV pressure control mode because it will result in further reduction of torus level.
C. The RCIC suction must remain aligned to the CST to prevent pump suction vortexing.
D. The RCIC system must be tripped to prevent over pressurizing the containment.

Proposed Answer: C. The RCIC suction must remain aligned to the CST to prevent pump suction vortexing. (OP-19 pp. 13)

Explanation (Optional):
A. The RCIC pump suction logic will not transfer on torus low level. This transfer occurs on CST low level.
B. RCIC operation in pressure control mode pumps water from the CST to the CST. OP-19 requires that the pump suction be aligned to the CST prior to starting RCIC in pressure control mode.
D. Containment can not be over pressurized by RCIC turbine exhaust. The containment vent will be able to "keep up" with the pressurization. (MIT-301.11E PP 11)

Technical Reference(s): OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: EK3. Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.5 / 45.6) EK3.03 RCIC operation: Plant-Specific 3.6 / 3.7

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	295034 EA1.01	
Importance Rating	_3.8_	_3.8_

Proposed Question: 40 / 30

The "A" reactor water cleanup (RWCU) pump has a seal leak. All reactor building temperature and radiation levels are normal except for the following. The RWCU pump area radiation monitor is reading 75 mr/hr and rising. The reactor building vent exhaust radiation monitors 17RM-452A & B are reading 3 x 10E3 and rising slowly. In addition to monitoring the RWCU pump area temperature and radiation levels what other actions must be taken.

A. Isolate the "A" RWCU pump and if reactor building vent radiation exceeds 1 x 10E4 then ensure SBGT starts and the reactor building has isolated.
B. Isolate the "A" RWCU pump, immediately isolate the reactor building and start SBGT.
C. Immediately isolate the reactor building and start the standby gas treatment system.
D. Enter EOP-2, "RPV Control."

Proposed Answer: A. Isolate the "A" RWCU pump and if reactor building vent radiation exceeds 1 x 10E4 then ensure SBGT starts and the reactor building has isolated.

Explanation (Optional):
B. EOP-5 does not require the vents to be isolated until 1X10E4 is reached and immediately isolating the RB vents will degraded the plant further because cooling to plant equipment will be further decreased.
C. EOP-5 does not require the vents to be isolated until 1X10E4 is reached and immediately isolating the RB vents will degraded the plant further because cooling to plant equipment will be further decreased.
D. Current conditions do not require this procedure to be entered.

Technical Reference(s): EOP-5, SDLP 12, 17, 01B

Proposed references to be provided to applicants during examination: EOP-5

Learning Objective: EA1. Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: (CFR: 41.7 / 45.6) EA1.01 Area radiation monitoring system 3.8 / 3.8

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295038 EK2.05	
Importance Rating	_3.7_	_4.7_

Proposed Question: 19 / 31

Which one of the following statements correctly describes the color code of the "OFFSITE RAD" box on the EPIC SPDS bar Display.

A. GRAY There are 3 or more invalid inputs in to the OFFSITE RAD" box.
B. GREEN There is an off site release at the ALERT level or lower emergency action level (EAL).
C. MAGENTA There is an off site release at the UNUSUAL EVENT level or higher EAL.
D. RED There is an off site release at the ALERT level or higher EAL.

Proposed Answer: D. RED There is an off site release at the ALERT level or higher EAL.

Explanation (Optional): This box is color coded red or green. Red if Alert or higher and Green is UE or lower.

Technical Reference(s): SDLP-66A, pp 61, EPlan

Proposed references to be provided to applicants during examination: None

Learning Objective: EK2. Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: (CFR: 41.7 / 45.8) EK2.05 †Site emergency plan 3.7 / 4.7*

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __7__
55.43 ______

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	600000 A2.16	
Importance Rating	_3.0_	_3.5_

Proposed Question: 43 / 32

The plant is operating at a 100% power, with a normal plant line up when a fire in the Control Room causes an immediate evacuation, such that none of the required immediate actions can be taken. All operator actions must be taken from outside the Control Room. What operator actions must be taken to scram the reactor and what other actions will result?

A. Both the RPS MG set output breaker AND RPS alternate feeder breaker must be OPENED on the A & B RPS system. This will result in a scram only.

B. The A & B RPS MG set output breakers must be OPENED. This will result in a scram only.

C. The A & B RPS MG set output breakers must be OPENED. This will result in a scram and group 1 isolation only.

D. The A & B RPS MG set output breakers must be OPENED. This will result in a scram, Group I isolation, and Group II isolation.

Proposed Answer: D. The A & B RPS MG set output breakers must be OPENED. This will result in a scram, Group I isolation, and Group II isolation.

Explanation (Optional):

A. Only the RPS MG Set breakers need be open to initiate a scram. Both breakers do not need to be opened.
B. This will result in more than a scram
C. If the RPS system losses power PCIS will actuate as well.
D. Correct.

Technical Reference(s):SDLP-16C, AOP-43 PLANT SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

Proposed references to be provided to applicants during examination: None

Learning Objective: AA2 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: AA2.12 Location of vital equipment within fire zone 3.1 / 3.5 A2.16 Vital equipment and control systems to be maintained and operated during a fire.

Question Source: Bank # FitzPatrick Requalification Question 126

A fire in the Control Room causes immediate evacuation, such that none of the required immediate actions can be taken. All actions must be taken from outside the Control Room. Which of the following statements is correct?

A. In order to scram the reactor the RPS A and B MG set output breakers AND RPS alternate feeder breakers must be opened.

B. De-energizing RPS A and B by opening both MG set output breakers will cause a scram ONLY.

C. De-energizing RPS A and B by opening both MG set output breakers will cause a scram, Group I isolation, and Group II isolation.

D. De-energizing RPS A and B by opening both MG set output breakers will cause a scram and Group I isolation ONLY.

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __8__
55.43 _____

Comments:

Examination Outline Cross-reference: Level	RO	SRO
Tier #	1	1
Group #	3	2
K/A #	295021	
Importance Rating	3.5	3.5

Proposed Question: 34 / 33

The plant is in shutdown with the "B" residual heat removal system operating in shutdown cooling mode. The following plant conditions exist.

RPV pressure ....0 psig
RPV level ....270 inches
RPV head ....INSTALLED
Coolant temperature ....100°F

"A" & "B" Reactor Water Recirculation (RWR) pumps ....OFF
"B" Reactor Recirculation Pump Suction Valve (02MOV-43B)...................OPEN
"B" Reactor Recirculation Pump Discharge Valve (02MOV-53B) ....CLOSED

Maintenance is troubleshooting the "B" Reactor Recirculation Pump Discharge Valve (02MOV-53B) when the valve inadvertently opens. What effect does this have on shutdown cooling.

A. This will create a large drain path to the torus and will reduce the RPV water level until the group II isolation occurs on RPV low water level.
B. This will result in a loss of shutdown cooling and insufficient natural circulation.
C. This will result in a reduction of shutdown cooling because some of the shutdown cooling flow will bypass the reactor core.
D. This will result in a loss of valid reactor coolant temperature indication because of insufficient natural circulation.

Proposed Answer: C. This will result in a reduction of shutdown cooling because some of the shutdown cooling flow will bypass the reactor core.

Explanation (Optional): Opening the RWR pump discharge valve will allow the RHR shutdown cooling flow to bypass the core. However, the RPV water level is greater than 234.5 which will ensure good natural circulation and temperature indication is available.

Technical Reference(s): SDLP-02H, SDLP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.7 / 45.6) AA1.02 RHR/shutdown cooling 3.5 / 3.5

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

Question History: Last NRC Exam ______
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 ______

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__3__	__1__
K/A #	295023 AA1.02	
Importance Rating	_2.9_	_3.1_

Proposed Question:11 / 34

The plant is shutdown for a refueling outage with the fuel pool gates installed. Annunciator 09-3-1-9 FUEL POOL COOL & CLN UP TROUBLE alarms. The NLO reports that the spent fuel pool level is slowly dropping and the running fuel pool cooling pump has tripped. Which one of the following methods is available to provide makeup to the spent fuel pool.

A. Align and inject core spray into the reactor cavity.
B. Start the second fuel pool cooling pump to refill the pool.
C. Align condensate transfer to makeup to the skimmer surge tanks.
D. Start a second control rod drive pump to inject into the reactor cavity.

Proposed Answer: C. Align condensate transfer to makeup to the skimmer surge tanks.

Explanation (Optional):
A. The fuel pool gates are installed and because of this, addition of water to the cavity will have no effect on the fuel pool level.
B. Fuel pool level has decreased and has resulted in the alarm. The level in the skimmer surge tanks are the same. The second pump will not start because the level in the skimmer surge tank has fallen below the low low level in the skimmer surge tank.
D. The control rod drive pump will inject into the reactor vessel and will not effect the level in the fuel pool.

Technical Reference(s):SDLP-19, AOP-53

Proposed references to be provided to applicants during examination: None

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: (CFR: 41.7 / 45.6) AA1.02 Fuel pool cooling and cleanup system 2.9 / 3.1

Question Source: Bank # INPO 6671

Which of the following methods/systems is normally used to refill the fuel storage pool?
From the "A" CST via the condensate transfer pumps and refill through the skimmer surge tank(s).

Align Shutdown Cooling and refill via the spent fuel pool diffusers.
Start the second FPCC pump and refill via the spent fuel pool diffusers
Cross connect with Unit 3 FPCC and refill via the spent fuel pool diffusers.

Reference: ..295023.A1.02

Question History:	Last NRC Exam	____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__3__	__2__
	K/A #	295035 A2.01	
	Importance Rating	_3.8_	_3.9_

Proposed Question: 41 / 35

During alignment of reactor building ventilation the operator notices that reactor building to atmosphere differential pressure is positive and rising. At what differential pressure will the Reactor Building ventilation isolate and what operator actions will be required to maintain secondary containment integrity?

A. At + 1 inch of water the reactor building isolation occurs, verify the standby gas treatment system has automatically started.
B. At + 1 inch of water the reactor building isolation occurs, manually start the standby gas treatment system.
C. At + 4 inches of water the reactor building isolation occurs, verify the standby gas treatment system has started.
D. At + 4 inches of water the reactor building isolation occurs, manually start the standby gas treatment system.

Proposed answer: B. At + 1 inch of water the reactor building isolation occurs, manually start the standby gas treatment system.

Explanation (Optional): At +1 inch water the reactor building isolates. Standby gas does not automatically start on this isolation so it must be manually started.

Technical Reference(s): OP-51A REACTOR BUILDING VENTILATION AND COOLING SYSTEM* and SDLP-01B & 66

Proposed references to be provided to applicants during examination: None

Learning Objective: EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.8 to 41.10) EA2.01 Secondary containment pressure: 3.8 / 3.9

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __9__
55.43 _____

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__3__	__2__
K/A #	295036 2.1.7	
Importance Rating	_3.7_	_4.4_

Proposed Question: 42 / 36

While the plant is operating at 100% power, a fault occurs in the fire protection system resulting in discharge into the east and west crescent areas. Water levels in both crescent areas are 20 inches and rising. Choose the statement below that describes the operator action that is required.

A. Scram the reactor.
B. Commence a reactor shut down.
C. Isolate sump discharge to radwaste storage tanks.
D. Perform an EMERGENCY DEPRESSURIZATION in accordance with EOP-2.

Proposed Answer: B. Commence a reactor shut down.

Explanation (Optional): The water levels in EOP-5, "Secondary Containment Control" have been exceeded and there is NOT a primary system discharging into the area, therefore a normal shutdown is directed.

Technical Reference(s): EOP-5, MIT 301.11F

Proposed references to be provided to applicants during examination: EOP-5

Learning Objective: 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation - RO 3.7/SRO 4.4
MIT-301.11F, EO 5.07

Question Source: Bank # FitzPatrick Requalification Bank 20005219B01C Rev.2
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __10_
55.43 _____
LO MIT-301.11F, EO 5.07

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__2__
	K/A #	201001 K2.05	
	Importance Rating	_4.5_	_4.5_

Proposed Question: 67 / 37

The alternate rod insertion (ARI) valve solenoids 03SOV-201 through 205 are powered from which one of the following sources?

A. 71 ACUPS
B. 71AC-9
C. The "A" 125 Volt DC Battery / Battery Charger
D. The "B" 125 Volt DC Battery / Battery Charger

Proposed Answers: C. The "A" 125 VDC Battery / Battery Charger

Explanation (Optional): Power to these valves are supplied through 71DC-A5, CKT #7. This is a distribution panel off of the "A" 125 VDC battery / battery charger.

Technical Reference(s): SDLP-03C, Table III power supplies.

Proposed references to be provided to applicants during examination: None

Learning Objective: K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)
K2.05 Alternate rod insertion valve solenoids: 4.5*/4.5*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__2__
K/A #	201002 K3.01	
Importance Rating	_3.4_	_3.4_

Proposed Question: 68 / 38

During weekly control rod drive testing the operator receives a reactor manual control timer malfunction. The timer malfunction can not be reset. How does the timer malfunction effect control rod movement?

A. Control rods can be inserted using the emergency in position of the emergency in/notch override switch.

B. Control rods can not be individually scrammed.

C. Control rods can not be selected to be moved because there is a select block

D. The rod worth minimizer blocks rod withdrawal because it loses rod position indication.

Proposed Answer: C. Control rods can not be selected to be moved because there is a select block

Explanation (Optional): Failure of the RMC timer will result in a select block. This will prevent a control rod from being selected and thus prevent any rod movement.

Technical Reference(s): SDLP-03F

Proposed references to be provided to applicants during examination: None

Learning Objective: K3. Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following: (CFR: 41.7 / 45.4)
K3.01 Ability to move control rods 3.4/3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__
	Group #	__1__	__
	K/A #	202002 A2.09	
	Importance Rating	_3.1__	_3.3_

Proposed Question: R 39

The plant is at 100% power. The "A" reactor recirculation pump scoop tube was "locked up;" however, while performing procedure OP-27, "Recirculation System," the operator failed to place the SCOOP TUBE AUTO UNLOCK control switch in ON. While the "A" reactor recirculation pump scoop tube was "locked up," the "B" feedwater pump trips. Based on these conditions __(1)__ and procedure __(2)__ will be used.

A. (1) Only the "B" reactor recirculation pump will run back to 44%
(2) AOP-1, "Reactor Scram."

B. (1) Both the A & B reactor recirculation pumps will run back to 44%
(2) AOP-42, "Feedwater Malfunction (Lowering Feedwater Flow)"

C. (1) Only the "A" reactor recirculation pump will trip
(2) AOP-8, "Loss or Reduction of Reactor Coolant Flow"

D. (1) Both the A & B reactor recirculation pumps will trip
(2) AOP-8, "Loss or Reduction of Reactor Coolant Flow"

Proposed Answer: A. (1) Only the "B" reactor recirculation pump will run back to 44%
(2) AOP-1, "Reactor Scram."

Explanation (Optional): Procedure OP-27, Recirculation System has the operator place the SCOOP TUBE AUTO UNLOCK control switch in ON. This will allow a locked up reactor recirculation pump to run back if a FW pump is lost. Since the operator did not place this in ON only one pump will run back. With only a single pump run back the reactor will scram on low level after "B" feedwater pump trip.

Technical Reference(s): OP-27, "Reactor Recirculation System, SDLP-021, SDLP-33

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.09 †Recirculation flow mismatch: Plant-Specific 3.1 / 3.3

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	___X__
10 CFR Part 55 Content:	55.41 __5___	
	55.43 __5___	

Comments:

Examination Outline Cross-reference: Level		RO	SRO
	Tier #	__2__	__2_
	Group #	__1__	__1__
	K/A #	202002 K5.01	
	Importance Rating	_2.8_	_2.8_

Proposed Question: 48 / 40

The plant is at power with both reactor recirculation (RWR) pumps operating at 90% speed. A significant oil leak occurs on the "A" RWR motor generator (MG) set piping. What effect will this have on the RWR system?

A. Both RWR pumps will automatically trip due to low oil pressure.
B. The "A" RWR pump scoop tube will lock up and the drive motor will trip on low oil pressure.
C. The "A" RWR pump scoop tube will lock up, allowing the oil in the fluid coupling reservoir to maintain the "A" RWR pump operating at a reduced speed.
D. The "A" RWR pump must be manually tripped.

Proposed Answer: B. The "A" RWR pump scoop tube will lock up and the drive motor will trip on low oil pressure.

Explanation (Optional): On low oil pressure the scoop tube will lockup and the drive motor breaker will trip, tripping the pump. The running pump flow will increase and sped must be reduced to less than 80% to prevent excessive jet pump differential pressure.

Technical Reference(s): SDLP-02H, AOP-8, Op-27

Proposed references to be provided to applicants during examination: None

Learning Objective: K5. Knowledge of the operational implications of the following concepts as they apply to RECIRCULATION FLOW CONTROL SYSTEM : (CFR: 41.5 / 45.3)
K5.01 Fluid coupling: BWR-3,4 - 2.8/2.8

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5___
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	203000 A1.04	
	Importance Rating	_3.6_	_3.6_

Proposed Question: 46 / 41

Several minutes ago a medium break loss of coolant accident occurred simultaneously with a loss of off-site power. The reactor scrammed, core spray (CS) pumps, and the residual heat removal (RHR) pumps automatically started on a valid initiation signal. Reactor vessel level is at 125 inches and reactor pressure is at 500 psig, both level and pressure are dropping slowly due to the leak. The "B" Loop of RHR was in a normal standby lineup before the event and currently has the following indications.

"B" RHR pump ....OPERATING
"D" RHR pump...OPERATING
"B" Loop RHR Flow 10FI-133 (09-3 panel) ....0 gpm
"B" & "D" RHR pump discharge pressure.............200 psig
"B" Loop RHR minimum flow valve (MOV-16B)..OPEN
"B" Loop RHR injection valve (MOV-27B)............OPEN
"B" Loop RHR injection valve (MOV-25B)............CLOSED

Assuming that reactor level and pressure continue to drop how will the "B" Loop of RHR respond.

A. The "B" & "D" RHR pumps will not provide sufficient flow because they have been operating without minimum flow several minutes.

B. The "B" Loop RHR injection valve (MOV-25B) has failed to OPEN which will prevent the "B" Loop of RHR from injecting unless the injection valve is opened locally.

C. The RHR injection valve (MOV-25B) will OPEN when reactor pressure reaches 450 psig, and when reactor pressure drops below 200 psig indicated flow will rise and the minimum flow valve will close.

D. The "B" Loop RHR injection valve (MOV-25B) will OPEN and the minimum flow valve (MOV-16B) will CLOSE simultaneously when reactor pressure reaches 450 psig, indicated flow will rise as the minimum flow valve closes.

Proposed Answer: C. The RHR injection valve (MOV-25B) will OPEN when reactor pressure reaches 450 psig, and when reactor pressure drops below 200 psig indicated flow will rise and the minimum flow valve will close.

Explanation (Optional):

- The shutoff head of the pumps are approximately 200 psig. The pumps are running on minimum flow. The only indication that the pumps are on minimum flow is the position of the minimum flow valve.
- The MOV-25B is normally closed and will open on an initiation signal and reactor pressure less than 450 psig.
- The MOV-27B is OPEN in the standby line up.

- The minimum flow valve will close on flow of 1450 gpm, not a reactor pressure of 450 psig

Technical Reference(s): SDLP-10, RHR

Proposed references to be provided to applicants during examination: None

Learning Objective: A1. Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: (CFR: 41.5 / 45.5) A1.04 System pressure 3.6 / 3.6

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_2_	_2_
	Group #	_1_	_1_
	K/A #	206000 K6.03	
	Importance Rating	_2.9_	_3.1_

Proposed Question: 47 / 42

A station blackout has occurred. What effect will the station blackout have on the HPCI system.

A. The HPCI system is not effected by a loss of AC power because there are no HPCI components that are powered by AC power.
B. The HPCI system will function; however, the flow controller will lose power and HPCI must be controlled locally.
C. The HPCI system will function; however, steam leakage along the shaft seal could occur.
D. The HPCI system will not start because the auxiliary oil pump will lose power and the turbine control valve will not open.

Proposed Answer: C. The HPCI system will function; however, steam leakage along the shaft seal could leak occur.

Explanation (Optional):
A. MOV 15 is powered from 600 VAC and will not automatically close on a HPCI isolation signal.
B. The HPCI flow controller will still have power it is powered from the "B" 125 VDC battery through an inverter.
D. The auxiliary oil pump is powered from DC.

Technical Reference(s): SDLP-23, HPCI

Proposed references to be provided to applicants during examination: None

Learning Objective: K6. Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM : (CFR: 41.7 / 45.7) K6.03 A.C. power: BWR-2,3,4 - 2.9 / 3.1*

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	209001 K4.01	
Importance Rating	_3.2_	_3.4_

Proposed Question: 49 / 43

Which one of the following statements correctly describes the automatic function of the core spray injection valves, 14MOV-11A(B) / 14MOV-12A(B)?

A. The core spray injection valves will open on a core spray automatic initiation signal when the core spray pump discharge pressure is 100 psig or greater to inject into the reactor vessel.

B. The core spray injection valves will open on a core spray automatic initiation signal when reactor pressure is less than 450 psig to protect the low pressure piping.

C. The core spray injection valves will automatically close at a reactor vessel level of 222.5 inches to prevent flooding and damaging the main steam lines.

D. The core spray injection valves will automatically close when the core spray sparger break detection logic is actuated to prevent flow diversion from the sparger.

Proposed Answer: B. The core spray injection valves will open on a core spray automatic initiation signal when reactor pressure is less than 450 psig to protect the low pressure piping.

Explanation (Optional): The core spray injections valves automatically open on an initiation signal and RPV pressure less than 450 psig. This protects the low pressure core spray piping. There are no automatic closures of the core spray injections valves on the RPV high level or sparger break detection logic.

Technical Reference(s): SDLP-14, CAD file S14-004.cdr

Proposed references to be provided to applicants during examination: None

Learning Objective: K4. Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: K4.01 Prevention of overpressurization of core spray piping 3.2 / 3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	211000 K2.01	
Importance Rating	_2.9_	_3.1_

Proposed Question: R44

An ATWS has occurred. Reactor power is at 25% and the scram discharge volume is full of water and pressurized. In addition, MCC-152, MCC-156, and MCC-166 have lost power. What actions must be taken to shut down the reactor.

A. Start SLC injection using the "A" SLC pump, the "B" SLC pump has lost power.
B. Start SLC injection using the "B" SLC pump, the "A" SLC pump has lost power.
C. Inject SLC Solution through the CRD System, both the "A" & "B" SLC pumps have lost power.
D. Vent the scram air header since both the "A" & "B" SLC pumps have lost power.

Proposed Answer: B. Start SLC injection using the "B" SLC pump, the "A" SLC pump has lost power.

Explanation (Optional):
A. MCC-152 inop "A" pump and MCC-162 powers the "B" pump. In this case the "A" pump has lost power.
B. Correct
C. The "B" SLC pump still has power.
D. Venting the scram air header will not result in further rod motion if there is a hydraulic lock. In addition the "B" pump still has power.

Technical Reference(s): SDLP-11, "SBLC" , TS 3.4

Proposed references to be provided to applicants during examination: None

Learning Objective: K2. Knowledge of electrical power supplies to the following (CFR: 41.7)
K2.01 SBLC pumps 2.9* / 3.1*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	212000 K1.15	
	Importance Rating	_3.8_	_3.9_

Proposed Question: 52 / 45

A reactor scram occurred 30 minutes ago. All plant systems responded as designed except for control rod 18-27. This control rod had a noticeable delay in starting to scram as compared to the other control rods. System engineering has performed a field inspection on 18-27 CRD hydraulic control unit (HCU) and determined that the "B" RPS scram pilot air valve did not reposition to scram the control rod. Based on this information, why did control rod 18-27 insert during the reactor scram?

A. Control rod 18-27 "A" RPS scram pilot air valve repositioned and bled the air off the scram outlet and inlet valves allowing them to open.
B. Control rod 18-27 "A" RPS scram pilot air valve repositioned and bled the air off the scram outlet valve allowing the valve to open and the rod to drift in.
C. Repositioning of scram pilot air valves on the other HCUs bled the scram air header down which resulted in control rod 18-27 scram outlet and inlet valves to open.
D. Repositioning of the backup scram valves bled the scram air header down allowing control rod 18-27 scram outlet and inlet valves to open.

Proposed Answer: D. Repositioning of the backup scram valves bled the scram air header down allowing control rod 18-27 scram outlet and inlet valves to open.

Explanation (Optional):

A. Both RPS scram pilot air valve must reposition to bled the air off the scram inlet and outlet valves.
B. Both RPS scram pilot air valve must reposition to bled the air off the scram inlet and / or outlet valves
C. Repositioning of the scram pilot air valve will block and prevent the scram air header from bleeding down.

Technical Reference(s): SDLP 05, Reactor Protection System

Proposed references to be provided to applicants during examination: None

Learning Objective: K1. Knowledge of the physical connections and/or cause- effect relationships between REACTOR PROTECTION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.15 SCRAM air header pressure 3.8/3.9

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X___
10 CFR Part 55 Content:	55.41 __2__	
	55.43 _____	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	215003 A3.04	
	Importance Rating	_3.5_	_____

Proposed Question: R46

The plant is at 35% power and performing a planned shutdown. The "B" intermediate range monitor (IRM) has failed upscale. The "B" IRM mode switch is in operate and has not been bypassed. When would you expect to receive a half scram and rod block from this IRM?

A. When the "A" average power range monitor (APRM) decreases to 5%.
B. When the reactor mode switch is moved to startup/Hot Standby
C. When the "B" IRM detector is fully inserted into the core.
D. As soon as the IRM failed upscale.

Proposed Answer: B. When the reactor mode switch is moved to startup/Hot Standby

Explanation (Optional): A. The "A" APRM is not the companion APRM for this IRM.
C. & D. The only time that an IRM will provide an automatic function is when the mode switch is not in RUN OR in RUN with the companion APRM (B) downscale.

Technical Reference(s): None

Proposed references to be provided to applicants during examination: None

Learning Objective: A3 Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: A3.04 Control rod block status 3.5/3.5

Question Source: Bank # INPO 8236

Given the following conditions:

-The plant is performing a scheduled shutdown
-Intermediate Range Monitoring (IRM) channel "B" has failed "UPSCALE" and has NOT been bypassed

At what point would an automatic half scram be expected for these conditions?

The plant enters Condition 2
APRM "B" reaches 5% power
IRM detectors are fully inserted
Power is below the Low Power Setpoint

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	215004 K2.01	
	Importance Rating	_2.6_	_2.8_

Proposed Question: R47

The system "A" 24/48 volt batteries (71IB-1, 71IB-2) and chargers (71IBC-1, 71IBC-2) supply the __(1)__ SRMs and the system "B" 24/48 volt batteries (71IB-3, 71IB-4) and charger (71IBC-3, 71IBC-4) supply the _(2)__ SRMs.

A. (1) A & B
(2) C & D

B. (1) A & C
(2) B & D

C. (1) A & D
(2) B & C

D. (1) B & D
(2) A & C

Proposed Answer: B. (1) A & C
(2) B & D

Explanation (Optional): The system "A" 24/48 VDC batteries / charger supplies SRM A & C and the system "B" 24/48 VDC batteries / charger supplies SRM B & D.

Technical Reference(s): SDLP-07B, SDLP-71B

Proposed references to be provided to applicants during examination: None

Learning Objective: K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)
K2.01 SRM channels/detectors 2.6/2.8

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

Question History: Last NRC Exam _____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __7__
55.43 __2__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	215005 K3.07	
Importance Rating	_3.2_	_3.3_

Proposed Question: 54 / 48

During a reactor startup the unit is at 35% power when the reactor operator selects the center control rod in preparation for withdrawal. The operator notices the following information on the 09-5 panel.

LPRM Status lights on the 4 Rod Display

Three DET A BYPASS lights lit
One DET B BYPASS light lit
Two DET C BYPASS lights lit
Zero DET D BYPASS lights lit

Based on this information what is the status of the RBM system.

A. The "A" RBM is automatically bypassed since there are too few inputs to the "A" RBM
B. The "B" RBM is automatically bypassed since there are too few inputs to the "B" RBM
C. The "A" RBM is providing a rod block to RMCS because there are too few inputs to the "A" RBM
D. The "B" RBM is providing a rod block to RMCS because there are too few inputs to the "B" RBM

Proposed Answer: C. The "A" RBM is providing a rod block to RMCS because there are too few inputs to the "A" RBM

Explanation (Optional): The RBM will provide a rod block signal to RMCS if there are to few LPRM inputs to the RBM circuitry. The RBM needs 50%, The "A" RBM does not have 50% of the LPRM inputs above the downscale values. The "A" uses the "A" and "C" level LPRMs and the "B" RBM uses the "B" and "D" LPRMs.

Technical Reference(s): SDLP-07C

Proposed references to be provided to applicants during examination: None

Learning Objective: K3. Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: (CFR: 41.7 / 45.4) K3.07 Rod block monitor 3.2/3.3

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 __7___	
	55.43 _____	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1___
K/A #	216000 K4.14	
Importance Rating	_3.3_	_4.3_

Proposed Question: 55 / 49

The narrow range RPV level indication is calibrated for _(1)_ , the wide range is calibrated for _(2)_ and the fuel zone level indication is calibrated for _(3)_ .

A. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 1000 psig / 546°F and 135°F drywell temperature
(3) RPV 0 psig / 212°F and 212°F drywell temperature

B. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 0 psig / 212°F and 212°F drywell temperature
(3) RPV 0 psig / 212°F and 212°F drywell temperature

C. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 1000 psig / 546°F and 135°F drywell temperature
(3) RPV 1000 psig / 546°F and 135°F drywell temperature

D. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 1000 psig / 546°F and 135°F drywell temperature
(3) RPV 0 psig / 212°F and 135°F drywell temperature

Proposed Answer: A. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 1000 psig / 546°F and 135°F drywell temperature
(3) RPV 0 psig / 212°F and 212°F drywell temperature

Explanation (Optional): Fuel Zone is used under accident conditions and is therefore calibrated under accident conditions. The narrow range and wide range are calibrated under normal conditions.

Technical Reference(s): SDLP-02B

Proposed references to be provided to applicants during examination: None

Learning Objective: K4. Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.14
Temperature compensation for reactor water level indication: 3.3/ 3.4

Question Source: Bank # ________
Modified Bank # _______ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	_2_	_2_
Group #	_1_	_1_
K/A #	217000 K3.02	
Importance Rating	_3.6_	_3.6_

Proposed Question: 50 / 50

A break has occurred in the steam supply line to the Reactor Core Isolation Cooling (RCIC) system upstream of the RCIC high flow sensing instrument taps (between the flow sensing taps and the reactor vessel). What effect will this have on RPV Pressure and RCIC?

A. RCIC will automatically isolate on high temperature in the drywell entrance area. The reactor will continue to depressurize.

B. RCIC will automatically isolate on high temperature in the drywell entrance area and stop the leak. Reactor pressure will be maintained by EHC and the turbine bypass valves.

C. RCIC will automatically isolate when sensed reactor pressure decreases to 50-100 psig. The reactor will continue to depressurize.

D. RCIC will automatically isolate and stop the leak when sensed reactor pressure decreases to 50-100 psig.

Proposed Answer: C. RCIC will automatically isolate when sensed reactor pressure decreases to 50-100 psig. The reactor will continue to depressurize.

Explanation (Optional): Based on SDLP-13, DWG# S13-012.cdr, a break upstream of the flow taps will not be isolated when a RCIC isolation occurs. In addition, the break is in the drywell and will not actuate the drywell entrance area temperature sensors. However, RCIC will isolate when reactor pressure decreases to 50-100 psig, but not isolate the leak.

Technical Reference(s): SDLP-13, LO 1.09

Proposed references to be provided to applicants during examination: None

Learning Objective: K3. Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: (CFR: 41.7 / 45.4) K3.02 Reactor vessel pressure 3.6 / 3.6

Question Source: Bank # INPO 7228
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis _X___

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Parent Question

QuestionId	7228	NSSSVendor	GE	CogLevel		Ka#	.217000.A2.15
		AbbrevLocName	Duane Arnold 1	ExamType	ILO		

Question:
A break has occurred in the steam supply line to the Reactor Core Isolation Cooling system upstream of the high flow sensing location. Which ONE of the following will provide system isolation antler this condition?

Reactor pressure low (50 psig)
RCIC emergency area cooler high temperature (175 deg F)
RCIC equipment room high vent inlet/outlet differential temperature (50 deg F)
Suppression pool area vent air high temperature (150 deg F)

Reference: ..217000.A2.15

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	217000 K5.07	
Importance Rating	_3.1_	_3.1_

Proposed Question: 56 / 51

Reactor Core Isolation Cooling (RCIC) has initiated due to a low RPV water level. When RPV water level reaches 222.5", RCIC steam supply isolation valve 13MOV-131 closes.

RCIC will reinitiate when RPV water level lowers to less than:

A. 222.5 inches.

B. 126.5 inches.

C. 222.5 inches, BUT the RCIC turbine trip/ throttle valve must be locally reset.

D. 126.5inches, BUT the RCIC turbine trip/ throttle valve must be locally reset.

Proposed Answer: B. 126.5 inches

Explanation (Optional): RCIC Turbine Steam Inlet Isolation Valve 13MOV-131 will close when RPV water level reaches 222.5 inches, the trip throttle valve will not close. The "131" will stay closed until the RPV water level lowers to 126.5 inches at which time RCIC will auto-initiate.

Technical Reference(s):OP-19 REACTOR CORE ISOLATION COOLING SYSTEM

Proposed references to be provided to applicants during examination: None

Learning Objective: K5. Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) : (CFR: 41.5 / 45.3) K5.07 Assist core cooling 3.1 / 3.1

Question Source: Bank # FitzPatrick Requalification 21701003B01C Rev. 2
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	218000 K6.05	
	Importance Rating	_3.0__	_3.1__

Proposed Question:57 / 52

A loss of the 71ACUPS-2 relay room uninteruptable bus distribution panel has just occurred. What effect will the loss of this distribution panel have on the automatic depressurization system (ADS)?

A. The ADS "A" initiation logic channel will lose power and automatically swap to the "B" initiation logic channel.
B. ADS has lost power to the pilot valve solenoids and will only actuate mechanically on high reactor pressure in the relief mode.
C. Red/Green light indication will be lost on the 09-4 panel.
D. The "white" open indication light above each control switch will not illuminate when the valve is open because the valve monitoring system has lost power.

Proposed Answer: D. The "white" open indication light above each control switch will not illuminate when the valve is open because the valve monitoring system has lost power.

Explanation (Optional):

A loss of UPS power only effects the VMS system. Logic power is supplied from 125 VDC

Technical Reference(s):SDLP-02J "ADS," OP-68 "ADS".

Proposed references to be provided to applicants during examination: None

Learning Objective: K6. Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM : (CFR: 41.7 / 45.7) K6.05 A.C. power: 3.0*/ 3.1*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7___
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	223001 A1.11	
Importance Rating	_3.1_	_3.2_

Proposed Question: 58 / 53

A loss of coolant accident is in progress, torus and drywell sprays have been initiated. Which one of the following will result if these sprays are NOT terminated and torus pressure continues to drop below 0.0 psig?

A. Chugging at the outlet of the downcomer will result in structural failure of the downcomers.
B. The reactor building to torus vacuum breakers will open at differential pressure of 0.5 psid and partially de-inert the Primary Containment.
C. The torus downcomer ring header will fail (collapse) at a differential pressure of 0.5 psid between the torus and drywell.
D. The Reactor Building to Torus vacuum breakers will fail at differential pressure of 0.5 psid

Proposed Answer: B. The reactor building to torus vacuum breakers will open at differential pressure of 0.5 psid and partially de-inert the Primary Containment.

Explanation (Optional):
A. Chugging will not occur because the drywell sprays are on which will take the non condensibles back into the drywell.
C. The downcomer will not collapse at this pressure.
D. These vacuum breakers are design to open at this pressure.

Technical Reference(s): SDLP-16A

Proposed references to be provided to applicants during examination: None

Learning Objective: Al. Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES controls including: (CFR: 41.5 / 45.5) A1.11 Reactor building to suppression chamber differential pressure: Plant-Specific 3.1/ 3.2

Question Source: Bank # INPO 303

A LOCA is in progress and drywell sprays have been initiated. Which one of the following will result if drywell sprays are NOT terminated and drywell pressure lowers below 0.0 psig?

Partial de-inerting of the Primary Containment.

Chugging at the outlet of the downcomer.
Mechanical failure (collapse) of the Torus downcomer ring header.
Mechanical failure of the Reactor Building to Torus vacuum breakers

Reference: ..226001.K3.01

Phenomenon associated with initiation of DW sprays.
Phenomen associated with evaporative cooling due to spraying while in the unsafe region of the DW spray initiation limit curve.
Event is within the design of the vacuum breakers.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	223002 A2.06	
	Importance Rating	_3.0_	_3.2_

Proposed Question: 59 / 54

The plant is operating at 100% power with no other activities in progress. If the A1 primary containment isolation system (PCIS) condenser vacuum instrument were to fail to 0 inches mercury vacuum, what effect would this have on the plant and what operator actions would be necessary?

A. The inboard "A" main steam line isolation valve (MSIV) will close, the reactor will scram on high APRM flux and the other MSIVs will close on high steam flow. Implement EP-9, "Opening MSIVs" to reopen the MSIVs.
B. The outboard "A" main steam line isolation valve (MSIV) will close, the reactor will scram on high APRM flux and the other MSIVs will close on high steam flow. Implement EP-9, "Opening MSIVs" to reopen the MSIVs.
C. All MSIVs would remain OPEN. Manually scram the reactor and CLOSE all MSIVs.
D. All MSIVs would remain OPEN. Verify that an isolation is not required and verify that the appropriate TS actions are implemented.

Proposed Answer: D. All MSIVs remain OPEN. Verify that an isolation is not required and verify that the appropriate TS actions are implemented.

Explanation (Optional):
A. These conditions would not have resulted in closure of the "A" MSIV. The logic requires 1 out of 2 taken twice.
B. These conditions would not have resulted in closure of the "A" MSIV. The logic requires 1 out of 2 taken twice.
C. A single instrument failure will not result in an MSIV closure.
D. Correct. Verify that an isolation is not required and enter appropriate TS. ARP 09-5-1-55. PCIS SYSTEM A ISOLATION

Technical Reference(s): SDLP 16C, PCIS and ARP 09-5-1-55. PCIS SYSTEM A ISOLATION

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.06 Containment instrumentation failures 3.0/ 3.2

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __5__	
	55.43 _____	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2___
	Group #	__1__	__1___
	K/A #	239002 A3.03	
	Importance Rating	_3.6_	_3.6_

Proposed Question: 60 / 55

Which of the following indications provide positive indication that a safety relief valve (SRV) is open.

A. Rise in the safety relief valve tail pipe temperature and white acoustic monitor light lit.
B. Safety relief valve red solenoid energized light lit and a rise in generator output.
C. Indicated rise in total core flow and white acoustic monitor light lit.
D. Indicated rise in total steam flow and drop in generator output.

Proposed Answer: A. Rise in the safety relief valve tail pipe temperature and white acoustic monitor light lit.

Explanation (Optional):
B. Generator output will decrease.
C. SRV open has no effect on total core flow (Comprehensive)
D. Indicated total steam flow decreases (Comprehensive)

Technical Reference(s): SDLP-71F and ARP-09-4-1-16

Proposed references to be provided to applicants during examination: None__

Learning Objective: A3. Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: (CFR: 41.7 / 45.7) A3.03 Tail pipe temperatures 3.6/3.6

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __7___
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	241000 A4.16	
Importance Rating	_3.3_	_3.2_

Proposed Question: 61 / 56

The plant is operating at 100% power when a transient occurs. You noticed a small upward spike on the APRMs and reactor pressure has taken a step rise of 3 psig and is constant at the new value. Reactor power remains at approximately 100%. Which one of the following indications, on the EHC console, is consistent with the event that has occurred.

A. "A" IN CONTROL light is ON
B. "B" IN CONTROL light is ON
C. LOAD LIMIT LIMITING light is ON
D. Mechanical Trip TRIPPED light is ON

Proposed Answer: B. EHC Console "B" IN CONTROL light is ON

Explanation (Optional): The indication of an APRM spike and then the step increase of 3 psig is the characteristic signature of a pressure regulator swap. The "A" pressure regulator is typically in control with the "B" pressure regulator set at 3 psig higher.

A. Since there has been no operator action to increase pressure and the pressure has increased by 3 psig the "A" in control light is OFF. It has failed in the upward direction.
C. This light is on when the load limiter is limiting the turbine load. This is set at 900 MWe (OP-9) if this were limiting load APRM would have increase significantly and stayed elevated.
D. If the mechanical trip light is ON the turbine has tripped. The turbine has not tripped.

Technical Reference(s):OP-9, AOP-6 and SDLP94C

Proposed references to be provided to applicants during examination: None

Learning Objective: A4. Ability to manually operate and/or monitor in the control room:(CFR: 41.7 / 45.5 to 45.8) A4.16 Lights and alarms 3.3/3.2

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	259001 A2.01	_____
	Importance Rating	_3.7_	_____

Proposed Question: R57

The plant is starting up at 80% power, 80% rod line with 96% reactor recirculation pump speed when the "A" reactor feedwater pump (RFP) trips. All systems operate as designed (after the pump trip). Which one of the following is the expected result of this trip and what operator actions are required?

A. The "B" RFP assumes the additional flow and reactor power is maintained at 80%. Monitor operation of the running feedwater, condensate & condensate booster pumps.

B. The third condensate & condensate booster pump starts and together with the "B" RFP assumes the additional flow and reactor power is maintained at 80%. Monitor operation of the running feedwater, condensate & condensate booster pumps.

C. The "B" RFP can not provide enough flow and a reactor scram will occur on low level. Place the reactor mode switch to shutdown and insert the intermediate range and source monitors.

D. Reactor recirculation flow and reactor power will drop and stabilize at approximately 50% power. Monitor for thermal-hydraulic instability and ensure that feedwater flow is within the normal capacity of the running pump.

Proposed Answer: D. The "B" RFP flow, reactor water recirculation flow and power will decrease and then stabilize at about 50% power. Monitor for thermal-hydraulic instability and ensure that feedwater flow is within the normal capacity of pump.

Explanation (Optional):
A. The "B" RFP does not have the capacity to maintain 80% power.
B. The third condensate/condensate booster pump does not provide additional capacity to the feedwater pump.
C. A reactor scram will not occur. All system will function as designed and the RWR will run back and plant stabilize at about 50% power.

Technical Reference(s): AOP-42 FEEDWATER MALFUNCTION (LOWERING FEEDWATER FLOW); AOP-1 REACTOR SCRAM*; AOP-8 LOSS OR REDUCTION OF REACTOR COOLANT FLOW*

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.01 Pump trip 3.7/3.7

Question Source: Modified Bank # FitzPatrick 25601012B02C Rev.3

The plant is operating at 65% power with all Condensate and Condensate Booster Pumps and both RFPs in service. There are no systems or components inoperable. The A Condensate Pump trips due to an electrical fault. Which one of the following is the expected result of this trip?

A. The operating pumps assume the additional load and the RFPs are not affected.
B. The A Condensate Booster Pump trips on interlock, but the RFPs are not affected.
C. The A Condensate Booster Pump trips on interlock causing RFPs to trip on low suction pressure.
D. Condensate Booster Pump suction pressure decreases causing RFPs to trip on low suction pressure.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	259002 2.4.34	
Importance Rating	_3.8_	_3.6_

Proposed Question: 62 / 58

A significant fire in the control room has resulted in the control room being evacuated. Which of the following indications should be used to obtain a value of reactor vessel water level from outside the control room in accordance with AOP-43, "Plant Shutdown From Outside the Control Room." (Actual reactor water level is 40 inches).

A. Wide range level indications on instrument rack 25-5
B. Narrow range level indications on instrument rack 25-5
C. Wide range level indications on instrument rack 25-6
D. Fuel Zone level indication on instrument rack 25-51

Proposed Answer: D. Fuel Zone level indication on instrument rack 25-51

Explanation (Optional): A&C. The wide range level indications can only be used for trend information and below 45 inches should not be used.
B. The Narrow range level indicator will be off scale for this level indication.

Technical Reference(s):AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.4.34 Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications. **RO 3.8/ SRO 3.6**

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	____
	Group #	__1__	____
	K/A #	261000 A3.01	
	Importance Rating	_3.2_	_3.3_

Proposed Question: R59

The plant is operating at 100% power with the "A" standby gas treatment (SBGT) system tagged out for maintenance. During maintenance activities on the refuel floor exhaust plenum radiation detectors, both detectors failed downscale and "B" SBGT started. The "B" SBGT system has the following indications 5 minutes after the system started.

B SBGT Fan ....RUNNING
SBGT Inlet Valve (14B)..............OPEN
SBGT Discharge Valve (15B).....OPEN
SBGT Flow ....1000 cfm

Based on these conditions the B SBGT system...

A. flow is in excess of the design flow rate for the system and will not achieve the desired hold time on the charcoal filters.
B. is operating at a reduced flow and may not be able to maintain the proper secondary containment pressure.
C. is operating as designed the system is operable.
D. should not have started, investigate why the B SBGT system started.

Proposed Answer: B is operating at a reduced flow and may not be able to maintain the proper secondary containment pressure.

Explanation (Optional):
A. The design flow rate for the system is 3000 - 6000 cfm.
B. The system flow is low and it may not be able to maintain the required secondary containment dP.
C. System flow is low.
D. The train should have started on this signal.

Technical Reference(s):OP-20 STANDBY GAS TREATMENT SYSTEM, SDLP-01B, SBGT

Proposed references to be provided to applicants during examination: None

Learning Objective: A3. Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: (CFR: 41.7 / 45.7) A3.01 System flow 3.2 / 3.3

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 __7___	
	55.43 _____	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	226001 A4.07	
	Importance Rating	_3.5_	_3.5_

Proposed Question: 63 / 60

The following plant conditions are present:

Torus pressure ....18.0 psig
RPV water level ....+30 inches indicated on the fuel zone level instruments
Drywell pressure and temperature ....Within the allowable region of the Drywell Spray Initiation Limit Curve

Which ONE of the following, by itself, will permit opening the Drywell spray valves for the 'A' RHR System?

A. Placing the DW & TORUS SPRAY VALVE OVERRIDE OF FUEL ZONE LEVEL Key lock Switch 10A-S18A in MANUAL OVERRIDE.
B. Closing the "A" LPCI Outboard Injection Valve 10MOV-27A.
C. Placing SPRAY CONTROL Switch 10A-S17A momentarily in RESET.
D. Placing SPRAY CONTROL Switch 10A-S17A momentarily in MANUAL.

Proposed Answer: D. Placing Containment Spray Control Switch 10A-S17A momentarily in MANUAL.

Explanation (Optional): The spray valves on the "A" loop will open if the following conditions are met:

1. Containment pressure >2.7 psig
2. SPRAY CONTROL switch is taken to the MAN position
3. RPV level is above 0 inches on the fuel zone instruments OR the DW & Torus Spray Valve Override Switch if taken to OVERRIDE.

A. This will not bypass any signal preventing valve opening if level is not less than TAF.
B. This will not allow the spray valves to be opened.
C. The switch must be taken to manual

Technical Reference(s):SDLP-10, pp53 and OP-13B

Proposed references to be provided to applicants during examination: None

Learning Objective: RHR / LPCI: Containment spray system mode A4 Ability to manually operate and/or monitor in the control room: (CFR 41.7/45.5 to 45.8) A4.07 Valve logic reset/ bypass/ override. 35. / 3.5

Question Source: Bank # FitzPatrick Requalification 0713.

The following plant conditions are present:

Torus pressure 18.0 psig
RPV water level +30 inches indicated by 02-3LI-91
Drywell pressure
and temperature Within the allowable region of the Drywell Spray Initiation Limit Curve

Which ONE of the following, by itself, will permit manual initiation of Drywell sprays from the 'A' RHR System?

E. Placing the DW & Torus Spray Valve Override of Fuel Zone Lvl Key lock Switch 10A-S18A in MANUAL OVERRIDE.
F. Closing the "A" LPCI Outboard Injection Valve 10MOV-27A.
G. Placing Containment Spray Control Switch 10A-S17A momentarily in RESET.
H. Placing Containment Spray Control Switch 10A-S17A momentarily in MANUAL.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	______	__2__
	Group #	______	__1__
	K/A #	264000 A2.01	
	Importance Rating	_3.5__	_3.6__

Proposed Question: 65 / 61

The "A" EDG is paralleled to the 10500 bus for performance of ST-9BA, "EDG A and C Full Load Test and ESW Pump Operability Test." While preparing to parallel "C" EDG, a LOCA occurs coincident with a loss of off site power. What effect would this have on the plant and what operator actions would be necessary?

A. The "A" EDG has been declared inoperable for the surveillance. The second RHR pump will automatically start, no operator action required.

B. The 10500 bus will not separate from the 10300 bus. The "C" EDG will automatically load to bus 10500. Operator action will be required to start a second RHR pump.

C. The 10500 bus will separate from the 10300 bus. The "C" EDG will not automatically close in to the 10500 bus. Operator action will be required to start a second RHR pump.

D. The 10500 bus will separate from the 10300 bus. The "C" EDG will automatically close in to the 10500 bus. The second RHR pump will automatically start, no operator action required.

Proposed Answer: C. The 10500 bus will separate from the 10300 bus. The "C" EDG will not automatically close in to the 10500 bus. Operator action will be required to start a second RHR pump.

Explanation (Optional):

A. The plant configuration will result in a loss of load shedding capability on the 10500 bus and manual starting of RHR pump is required.
B. The "C" EDG will not load onto this bus because the "A" EDG will be able to maintain normal voltage at the bus.
D. The "C" EDG will not load onto this bus because the "A" EDG will be able to maintain normal voltage at the bus.

Technical Reference(s): ST-9BA, SDLP-93 (pp45, 46, 69), OP-22

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.01 Parallel operation of emergency generator 3.5/3.6

SDLP-93 1.05.b.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __6__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	_____
Group #	__1__	_____
K/A #	218000 A4.07	
Importance Rating	_3.5_	_3.8_

Proposed Question: R62

The plant is operating at 100% power. Three minutes ago the white light above the control switch for the "A" SRV came ON. The following plant conditions are present.

Torus Temperature ....85° F and steady
Torus Level ....13.9 feet and steady
Total Steam Flow ....10.97 Mlbm/hr
Total Feedwater Flow..................10.97 Mlbm/hr

What has occurred to the "A" SRV?

A. The SRV is OPEN based on the white light indicating the pilot solenoid valve is open.
B. The SRV is OPEN based on the white light indicating high tail pipe temperature.
C. The SRV is CLOSED and the "A" SRV acoustic monitor has failed.
D. The SRV is CLOSED and the uninterruptible power supply has lost power.

Proposed Answer: C. The SRV is CLOSED and the "A" SRV acoustic monitor has failed.

Explanation (Optional):
A. The SRV is not open. There is no FF/SF mismatch.
B. The SRV is not open and the white light is acoustic monitor not temperature.
C. The acoustic monitor has failed and the SRV is closed.
D. This light will not illuminate on a loss of UPS.

Technical Reference(s): SDLP-02J, AOP-36, SDLP-06

Proposed references to be provided to applicants during examination: None

Learning Objective: A4. Ability to manually operate and/or monitor in the control room: A4.07 ADS valve acoustical monitor noise: 3.5/3.8

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

Question History: Last NRC Exam _____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 _____
55.43 _____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	201001 2.1.7	
	Importance Rating	_3.7_	_____

Proposed Question: R63

Control rods are being withdrawn in accordance with the sequence. When control rod 26-27 is withdrawn from position 12 to 24 the following control rod drive hydraulic system flow indications occur. The system flow was constant at 60 gpm prior to control rod movement. As the control rod is moved the system flow oscillates abnormally. After the control rod settles at position 24 the system flow steadies out at 60 gpm. Based on these indications what component failure has occurred in the control rod drive system.

A. The scram outlet valve for this control rod has failed OPEN.
B. The control rod drive flow control valve has failed OPEN.
C. The in-service stabilizing valve fail to CLOSED.
D. The insert directional control valve 122 has failed CLOSED.

Proposed Answer: C. The in-service stabilizing valves fail CLOSED.

Explanation (Optional):
A. If the scram outlet valve failed OPEN then the control rod would drift into the core.
B. If the valve failed open then the flow would be much higher than 59 gpm and would stay elevated.
D. If the 122 valve failed closed then the rod would not be able to move.

Technical Reference(s): SDLP03C; OP-25

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation.3.7/ 4.4

LO 1.05.a.6

Question Source: Bank # _______
Modified Bank #FITZPATRICK 20101011CRDC012 Rev.1

Assume the plant is operating at 100% steady state when the inservice CRDH stabilizing valves fail closed.

The CRDH system flow will:

a) Drop by approximately 6 gpm and remain steady during control rod movements.

b) Rise by approximately 4 gpm, but would oscillate during control rod movements.

c) Remain unchanged and would remain steady during control rod movements.

d) Remain unchanged, but would oscillate during control rod movements.

Answer D

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__6__
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__1__	_____
	K/A #	215004 K1.06	
	Importance Rating	_2.8_	_____

Proposed Question: R64

Which one of the following components forms the source range monitoring (SRM) system pressure boundary, extends through the core and is anchored in the upper core grid.

A. Drive Tube
B. Shuttle Tube
C. Dry Tube
D. Guide Tube

Proposed Answer: C. Dry Tube

Explanation (Optional):

A. The drive tube engages the motor and moves the detector / shuttle tube. This is inside the dry tube.

B. The shuttle tube moves inside the dry tube and has the detector attached. This tube moves up and down on top of the drive tube.

C. The dry tube forms the pressure boundary of the reactor vessel and is welded at the bottom of the in-core instrument housing and goes to the upper core grid and is anchored at the upper core grid.

D. In-core guide tube is located below the core plate area and protects the dry tube from flow induced vibrations.

Technical Reference(s): SDLP-07B, 02A.

Proposed references to be provided to applicants during examination: None

Learning Objective: K1. Knowledge of the physical connections and/or cause- effect relationships between SOURCE RANGE MONITORING (SRM) SYSTEM and the following: K1.06 Reactor vessel 2.8/ 2.8

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 _3, 7_
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	_____
Group #	__2__	_____
K/A #	201003 K4.02	_____
Importance Rating	_3.8_	_____

Proposed Question: R65

Control rod 34-03 is uncoupled from the control rod drive. If the control rod were to be selected and withdrawn, using notch override, from position 42 to position 48 what alarms and indications would be expected?

A. As soon as the control rod is withdrawn from position 42 the ROD DRIFT annunciator will alarm and when the rod motion control switch is released the control rod will drift out past position 48 and the rod position indication will be blank.

B. As soon as the control rod is withdrawn from position 42 the ROD DRIFT annunciator will alarm and the control rod will drift out to position 48. When the control rod position indicates 48 the ROD OVERTRAVEL annunciator alarms.

C. When the control rod is withdrawn past position 48 the ROD OVERTRAVEL annunciator alarms and after the rod motion control switch is released the ROD DRIFT annunciator will alarm and the rod position indication will be blank.

D. When the control rod is withdrawn past position 48 the ROD OVERTRAVEL annunciator alarms and after the rod motion control switch is released the ROD DRIFT annunciator will alarm and the rod position indication will indicate 48.

Proposed Answer: C. When the control rod is withdrawn past position 48 the ROD OVERTRAVEL annunciator alarms and after the rod motion control switch is released the ROD DRIFT annunciator will alarm and the rod position indication will be blank.

Explanation (Optional):

A. The control rod drift annunciator will not alarm as long as there is a drive signal present. In addition, the control rod will not drift. The drive will function as normal between 42 and 46. At position 48 the control rod will go beyond 48 to position 49 and receive the overtravel annunciator.

B. The rod overtravel annunciator alarms after the rod has gone past 48, to pick up reed switch 49.

C. Correct

D. The control rod position will be blank not indicating 48.

Technical Reference(s): AOP-25

Proposed references to be provided to applicants during examination: None

Learning Objective: K4. Knowledge of CONTROL ROD AND DRIVE MECHANISM design feature(s) and/or interlocks which provide for the following: K4.02 Detection of an uncoupled rod 3.8/ 3.9

Question Source:

Bank # ________

Modified Bank # ________ (Note changes or attach parent)

New ___X___

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_2_	_2_
	Group #	_2_	_2_
	K/A #	201006 K1.02	
	Importance Rating	_3.4_	_3.4_

Proposed Question: 69 / 66

A startup is in progress at 30% power. In-sequence control rod 26-27 has just settled at its target out position of 24. Prior to selecting the next control rod, control rod 26-27 loses position indication at position 24. Which one of the statements below is an acceptable operator response to this condition.

A. No action is required because the rod worth minimizer (RWM) control rod blocks are disabled above the low power set point.

B. Bypass the RWM and insert control rod 26-27 to the RWM alternate position of 22.

C. No actions is required because the RWM will only apply insert blocks above the low power set point.

D. Insert a substitute position of 24 for control rod 26-27 and re-initialize the RWM to clear the rod block.

Proposed Answer: B. Bypass the RWM and insert control rod 26-27 to the RWM alternate position of 22.

Explanation (Optional): The loss of RPIS for a single control rod or the total loss of a RPIS will result in a RWM program abort. This will result in insert and withdrawal blocks from 0% power through 35% power. RWM rod blocks are automatically bypassed above the low power set point; however, program aborts are not. Since control withdraw is continuing AOP-26, Loss of RPIS will not allow the operator to insert a substitute position on a control rod.

Technical Reference(s): SDLP-03D, Rod Worth Minimizer, OP-64, Rod Worth Minimizer.

Proposed references to be provided to applicants during examination: None

Learning Objective: K1. Knowledge of the physical connections and/or cause- effect relationships between ROD WORTH MINIMIZER and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.02 RPIS: 3.4 / 3.4

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis __x__

10 CFR Part 55 Content: 55.41 _2-9_
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	202001 K5.10	
Importance Rating	__2.8_	_2.8__

Proposed Question: 70 / 67

IF the speed controller for the first RWR pump MG Set to be placed in service was left at 50% prior to placing the MG set control switch to START, the MG set would:

A. Trip on overcurrent (50/51 relays).
B. Start and speed would rise to approximately 45%.
C. Overspeed because the generator field breaker would not close.
D. Start, speed would rise to 80%, then return to approximately 26%.

Proposed Answer: D. Start, speed would rise to 80%, then return to approximately 26%.

Explanation (Optional):

Technical Reference(s):See Above

Proposed references to be provided to applicants during examination: None

Learning Objective: K5. Knowledge of the operational implications of the following concepts as they apply to RECIRCULATION SYSTEM : (CFR: 41.5 / 45.3) K5.10 Motor generator set operation: Plant-Specific 2.8*/ 2.8

Question Source: Bank # FitzPatrick Requal No.20201004B03C Rev. 2

IF the speed controller for the first RWR pump MG Set to be placed in service was left at 50%, prior to placing the MG set control switch to START, the MG set would:

A. Trip on overcurrent (50/51 relays).
B. Start and speed would rise to approximately 45%.
C. Overspeed because the generator field breaker would not close.
D. Start, speed would rise to 80%, then return to approximately 26%.

Answer D

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __x___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5___
55.43 _____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__2__	__2__
	K/A #	204000 K6.05	
	Importance Rating	_2.6_	_2.6_

Proposed Question: 71 / 68

Maintenance has de-energized MCC-152 for planned maintenance. How does this maintenance effect the RWCU system?

A. The "A" RWCU pump (P-1A) will no longer have power available to the motor.
B. The "B" RWCU pump (P-1B) will no longer have power available to the motor.
C. The RWCU system outboard supply isolation valve (MOV-18) will lose power, and the supply line will have to be isolated in accordance with Technical Specifications.
D. The RWCU system inboard supply isolation valve (MOV-15) will lose power, and the supply line will have to be isolated in accordance with Technical Specifications.

Proposed Answer: D. The RWCU system supply isolation valve (MOV-15) will loose power, and the supply line will have to be isolated in accordance with Technical Specifications.

Explanation (Optional):
A. Supplied from MCC-131
B. Supplied from MCC-141
C. Supplied from "B" 125 VDC

Technical Reference(s): SDLP 12

Proposed references to be provided to applicants during examination: None

Learning Objective: K6. Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER CLEANUP SYSTEM : (CFR: 41.7 / 45.7) K6.05 A. C. power 2.6/2.6

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __7___
55.43 ______

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	_____
Group #	__2__	_____
K/A #	205000 K2.02	
Importance Rating	_2.5_	_2.7_

Proposed Question: R69

The "B" loop of the residual heat removal (RHR) system has been operating in shutdown cooling for the last 3 hours. Maintenance has requested to take Bus 11500 out of service for inspection. What effect if any, will this have on the "B" loop shutdown cooling valves.

A. This will have no effect. There are no "B" loop shutdown cooling valves that are effected by loss of Bus 11500.
B. This will disable the "B" RHR heat exchanger bypass valve (MOV-66B). This valve must be manually positioned to maintain the desired RPV water temperature.
C. This will disable the shutdown cooling suction outboard isolation valve (MOV-17) and prevent isolation of the penetration on a valid isolation signal.
D. This will disable the shutdown cooling suction inboard isolation valve (MOV-18) however it will not prevent isolation of the penetration on a valid isolation signal.

Proposed Answer: D. This will disable the "B" loop shutdown cooling suction inboard isolation valve (MOV-18) however it will not prevent isolation of the penetration on a valid isolation signal.

Explanation (Optional):
A. Bus 11500 feeds MCC 156 which feeds the inboard SDC isolation valve.
B. This valve is feed from bus 11600 / MCC
C. The outboard valve is DC

Technical Reference(s): SDLP - 10 RHR and SDLP-710 Figure 1

Proposed references to be provided to applicants during examination: None

Learning Objective: K2 Knowledge of electrical power supplies to the following: K2.02 Motor operated valves 2.5*/ 2.7*

SDLP-10, EO 1.04.b, power supplies for the shutdown cooling containment suction valves.

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	214000 2.2.24	
Importance Rating	_2.6_	_3.8_

Proposed Question: 72 / 70

The plant is at 100% power and maintenance has requested to remove the RPIS buffer card for control rod 06-31 which is currently at position 48. This will remove all position indication for this control rod. What actions, if any must be performed to release this maintenance activity?

A. A single rod scram time test must be performed.
B. A control rod coupling check must be performed.
C. The control rod must be declared inoperable.
D. No actions are necessary.

Proposed Answer: C. The control rod must be declared inoperable.

Explanation (Optional): A.&C. These tests are not required by TS for this activity
D. The control rod must be declared inoperable and electrically disarmed by TS 3.3.A.2b&d.

Technical Reference(s):AOP-26 LOSS OF ROD POSITION INDICATION*; TS 3.3.A.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.2.24 Ability to analyze the affect of maintenance activities on LCO status. RO 2.6/SRO 3.8

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__2__	__2__
	K/A #	215002 A2.03	
	Importance Rating	_3.1_	_3.3_

Proposed Question: 73 / 71

The "C" average power range monitor (APRM) failed downscale, all other APRMs are indicating 100% power with no abnormal indications. What effect will this failure have on the RBM system and what actions must be taken to correct this condition.

A. The "A" RBM will generate a rod block. The "C" APRM and "A" RBM must be manually bypassed to clear the rod block.
B. The "A" RBM will generate a rod block. The "A" RBM must be manually bypassed to clear the rod block.
C. The "A" RBM will automatically be bypassed. The "C" APRM must be manually bypassed which places the "E" APRM in service as the "A" RBM reference APRM.
D. The "A" RBM will automatically be bypassed. The "C" APRM must be manually bypassed which places the "F" APRM in service as the "A" RBM reference APRM.

Proposed Answer: C. The "A" RBM will automatically be bypassed. The "C" APRM must be manually bypassed which places the "E" APRM in service as the "A" RBM reference APRM.

Explanation (Optional): A.& B. The RBM does not generate a rod block, it is automatically bypassed when the reference APRM is less than 30 power.
D. The correct reference APRM is E not F.

Note: The RBM will generate a rod block; however, since the reference APRM is downscale the RBM is automatically bypassed and the rod block is not passed to the RMCS.

Technical Reference(s):SDLP-07C

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.03 Loss of associated reference APRM channel: 3.1/3.3

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 __5___
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	_____
Group #	__2__	_____
K/A #	219000 K6.02	
Importance Rating	_2.5_	_____

Proposed Question: R72

The "A" loop of residual heat removal (RHR) system has been operating in torus cooling with the "A" RHR pump at 7500 gpm. A moment ago annunciator RHR A LOGIC POWER FAILURE alarmed. What effect does this have on the "A" loop of the RHR system?

A. The "A" RHR pumps trips and the "A" RHR loop pressure drops to less than 90 psig.
B. The RHR test & torus cooling valve (10MOV-34A) will automatically close and the RHR loop pressure will increase to approximately the pump shutoff head pressure.
C. The RHR minimum flow valve (10MOV-16A) will close.
D. The "A" loop of RHR will continue to run in torus cooling. The pump minimum flow valve will not automatically open.

Proposed Answer: D. The "A" loop of RHR will continue to run in torus cooling. The pump minimum flow valve will not automatically open.

Explanation (Optional):
A. The RHR will not trip on loss of logic power. It will not be able to be remotely tripped or started.
B. The 34A valve will not close and in addition the automatic isolation has been lost for this valve.
C. The RHR minimum flow will already be closed because flow is at 7500 g.p.m. (>1250gpm).
D. AOP-22, states that the minimum flow valve will go closed. In this case the minimum flow valve is already closed and therefore, will not open.

Technical Reference(s):AOP-22, OP-13B, ARP 09-3-1-23

Proposed references to be provided to applicants during examination: None

Learning Objective: K6.Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI:TORUS/SUPPRESSION POOL COOLING MODE: K6.02 D.C. electrical power 2.5*/ 2.8*

SDLP-71B, Objective 1.09.a.5, Given a set of plant conditions, describe the effect that a loss of each of the DC electrical systems may have on the RHR system.

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 __7__	
	55.43 ______	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__3__
K/A #	239001 A2.07	
Importance Rating	_3.8_	_3.9_

Proposed Question: 82 / 73

The plant is operating at 85% power when a transient results in fuel failure and a small main steam line (MSL) leak in the steam tunnel. What is the expected plant response and what operator actions will be required after the automatic actions have completed?

A. High MSL flow will result in a reactor scram and high MSL temperature will close the main steam isolation valves (MSIVs). Implement actions in AOP-1, Scram and EOP-2, RPV control.

B. High MSL radiation will result in a reactor scram and high steam tunnel temperature will close the MSIVs. Implement actions in AOP-1, Scram and EOP-4, Primary Containment Control.

C. High steam tunnel temperature will close the MSIVs and the reactor will scram on the MSIV valve position. Implement actions in AOP-1, Scram and EOP-2, RPV control.

D. High steam tunnel temperature will close the MSIVs and the reactor will scram on high reactor vessel pressure. Implement actions in AOP-1, Scram and EOP-4, Primary Containment Control .

Proposed Answer: C. High steam tunnel temperature will close the MSIVs and the reactor will scram on the MSIV valve position. Implement actions in AOP-1, Scram and EOP-2, RPV control.

Explanation (Optional):

A. High MSL flow does not result in a scram.
B. The MSL high radiation scram / isolation has been disabled.
C. Correct.
D. The reactor will not scram on high reactor vessel pressure. The reactor will scram on MSIV valve position.

The MSIV will close from high steam tunnel temperature and the closure of the MSIVs will result in a scram and high vessel pressure. Primary containment control will be needed as soon as the MSIVs close due to the SRV lifting.

Technical Reference(s): SDLP-16C

Proposed references to be provided to applicants during examination: EOPs

Learning Objective: A2. Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.07 Main steam area high temperature or differential temperature high 3.8/3.9

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level		RO	SRO
		Tier #	__2__	__2__
		Group #	__2__	__2__
		K/A #	245000 A3.06	
		Importance Rating	_2.5_	_2.6_

Proposed Question: 74 / 74

The Unit has just experienced a turbine trip from 100% power and the turbine is currently costing down. Which one of the following describes the expected response of turbine support systems as the main turbine coasts down?

A. The Motor Suction pump and Emergency Bearing Oil pump will automatically start to supply the Main Turbine oil requirements.
B. The Motor Suction pump and Turning Gear Oil pump will automatically start to supply the Main Turbine oil requirements.
C. ONLY the Emergency Bearing Oil pump will start to supply the Main Turbine oil requirements.
D. NONE of the motor driven oil pumps will start.

Proposed Answer: B. The Motor Suction pump and Turning Gear Oil pump will automatically start to supply the Main Turbine oil requirements.

Explanation (Optional): SDLP-94A, states that during a turbine coast down the TGOP and MSP will automatically start. Both of these pumps start on low oil pressure.

Technical Reference(s):SDLP-94A pp. 52

Proposed references to be provided to applicants during examination: None

Learning Objective: A3. Ability to monitor automatic operations of the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS including:(CFR: 41.7 / 45.7) A3.06 Turbine lube oil pressure 2.5/2.6

Question Source: Bank # FitzPatrick Requal Exam Bank Question 0830
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	_____
Group #	__2__	_____
K/A #	256000 K3.04	
Importance Rating	_3.6_	_3.7_

Proposed Question: R75

The plant is operating at 65% power with all Condensate and Condensate Booster Pumps and both RFPs in service. There are no systems or components inoperable. The A Condensate Pump trips due to an electrical fault. Which one of the following is the expected result of this trip?

A. The operating pumps assume the additional load and the RFPs are not affected.
B. The A Condensate Booster Pump trips on interlock, but the RFPs are not affected.
C. The A Condensate Booster Pump trips on interlock causing RFPs to trip on low suction pressure.
D. Condensate Booster Pump suction pressure drops causing RFPs to trip on low suction pressure.

Proposed Answer: A. The operating pumps assume the additional load and the RFPs are not affected.

Explanation (Optional):

Technical Reference(s): N/A

Proposed references to be provided to applicants during examination: None

Learning Objective: K3. Knowledge of the effect that a loss or malfunction of the REACTOR CONDENSATE SYSTEM will have on following: K3.04, Reactor Feedwater System (3.6/3.7)

SDLP-33, EO 1.05.b.2 & 1.14.c

Question Source: Bank # FITZPATRICK 25601012B02C Rev.3

The plant is operating at 65% power with all Condensate and Condensate Booster Pumps and both RFPs in service. There are no systems or components inoperable. The A Condensate Pump trips due to an electrical fault. Which one of the following is the expected result of this trip?

a) The operating pumps assume the additional load and the RFPs are not affected.
b) The A Condensate Booster Pump trips on interlock, but the RFPs are not affected.
c) The A Condensate Booster Pump trips on interlock causing RFPs to trip on low suction pressure.
d) Condensate Booster Pump suction pressure decreases causing RFPs to trip on low

suction pressure.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	__ __

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__1__
K/A #	2620001 A3.02	
Importance Rating	_3.2_	_3.3_

Proposed Question: 64 / 76

The plant has just experienced a turbine trip / reactor scram and a fault on the 10400 bus. A residual transfer of electrical buses occurred. Which one of the following correctly describes the resultant configuration of the electrical buses?

A. Buses 10100, 10200, 10300, 10400 and 10700 de-energized
Buses 10300, 10400 10500 and 10600 energized from reserve power

B. Buses 10100 and 10200 de-energized;
Buses 10300, 10400 and 10700 energized from reserve power;
Buses 10500 and 10600 energized from either reserve power or EDGs

C. Buses 10400 and 10700 de-energized;
Buses 10100, 10200, 10300 energized from reserve power;
Bus10600 energized from either reserve power or EDGs
Bus10500 energized from EDGs

D. Buses 10100, 10200, 10400 and 10700 de-energized;
Bus 10300 energized from reserve power;
Bus10500 energized from either reserve power or EDGs
Bus10600 energized from EDGs

Proposed Answer: D. Buses 10100, 10200, 10400 and 10700 de-energized;
Bus 10300 energized from reserve power;
Bus10500 energized from either reserve power or EDGs
Bus10600 energized from EDGs

Explanation (Optional): Since Bus 10400 has faulted this bus will be dead and bus 10600 will be powered from the EDGs.

OP-22 Under normal operating conditions, the normal AC service power source and the off-site reserve AC power source are available to supply each emergency bus. The loss of the normal plant service power source results in automatic fast transfer to the off-site reserve AC power source. In the event of a failure to fast transfer, an automatic residual transfer will be made after a 3 second time delay, and each emergency diesel generator (EDG) will auto-start.

AOP-57 reserve supply breakers 10312 and 10412 close after bus voltage has been less than 25 percent of rate for greater than 3 seconds after the normal supply breaker trip.

Technical Reference(s): OP-46A, AOP-57, SDLP-71E

Proposed references to be provided to applicants during examination: None

Learning Objective:	A3. Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: (CFR: 41.7 / 45.7) A3.02 Automatic bus transfer 3.2/3.3
	EO 1.03, 1.09, 1.14
Question Source:	Bank # LO331E~1.doc question 16 Modified Bank # A faulted bus 10400 was added to the question New ________
Question History:	Last NRC Exam ________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__2__	__2__
	K/A #	262002 A4.01	
	Importance Rating	_2.8_	_3.1_

Proposed Question: 75 / 77

The uninterruptible power supply (UPS) is being supplied by the alternate AC supply. Work has been completed and the UPS is ready to be powered from the preferred UPS source (motor generator set). What operator action, by procedure, must be taken in the control room to support transferring the UPS back to normal power?

A. Take manual control of both reactor feedwater pumps by lowering the MSC.
B. Lock up the "A" & "B" reactor water recirculation (RWR) scoop tubes.
C. Isolate the reactor water cleanup system to prevent an automatic isolation.
D. Drive the SRM and IRM detectors into the core.

Proposed Answer: B. Lock up the "A" & "B" reactor water recirculation (RWR) scoop tubes.

Explanation (Optional):
A. If there is a complete loss of the UPS ONE FW pump will be controlled using the MSC, not both.
B. This action is correct based on OP-27. All other actions are taken locally at the UPS panel.
C. This action is not required by procedure, only a verification of the isolation if a complete loss of the UPS occurs.
D. The SRMs and IRMs are not procedurally required to be driven into the core.

Technical Reference(s): AOP-21, OP-46b, OP-27 and SDLP-71F

Proposed references to be provided to applicants during examination: None

Learning Objective: A4. Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.01 Transfer from alternative source to preferred source 2.8/3.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__2__	__2__
	K/A #	263000 2.4.10	
	Importance Rating	_3.0_	_3.1_

Proposed Question: 76 / 78

The plant is operating at 100% power when annunciator 09-8-4-23, "EDG D CNTRL PWR LOSS," alarms. What effect does this condition have on the "D" emergency diesel generator (EDG) if a valid start signal were present?

A. 120 VAC UPS control power has been lost. The EDG will not auto-start.
B. 120 VAC UPS control power has been lost. The EDG will auto-start.
C. 125 VDC control power has been lost. The EDG will not auto-start.
D. 125 VDC control power has been lost. The EDG will auto-start.

Proposed Answer: C. 125 VDC control power has been lost. The "D" EDG will not auto-start.

Explanation (Optional): ARP 09-8-4-23, states that the loss of 125 VDC to the control circuit results in the loss of EDG D starting and shutdown capability.

Technical Reference(s): ARP 09-8-4-23

Proposed references to be provided to applicants during examination: None

Learning Objective: 263000 DC Electrical Distribution 2.4.10 Knowledge of annunciator response procedures. RO 3.0/ SRO 3.1

Question Source: Bank # INPO
Modified Bank # 7567

Loss of all 125 VDC power would have which one of the following effects on the Emergency Diesel Generators?

Answer Diesels would not start and could not be started at the Engine Control Panel

Diesels would auto-start and load.
Diesels would auto-start, but would not auto load.
Diesels would not auto-start but could be started at the Engine Control Panel.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	_____
	Group #	__2__	_____
	K/A #	272000 K1.10	
	Importance Rating	_3.4_	_3.6_

Proposed Question: R79

The reactor building ventilation system has isolated on high radiation levels on the refueling floor (17RM-456A & B). All systems have functioned as designed. Which one of the following best describes the effects on primary and secondary containment?

A. 1. Reactor Building automatically Isolates and Recirculation Dampers OPEN
2. Standby gas treatment (SBGT) system must be manually started
3. Components in CAM, PCP, and H2/O2 systems isolate

B. 1. Reactor Building automatically Isolates and Recirculation Dampers CLOSE
2. SBGT system must be manually started
3. Refueling Floor Fans Trip

C. 1. Reactor Building automatically Isolates and Recirculation Dampers OPEN
2. SBGT system automatically starts
3. Refueling Floor Exhaust Fans Trip

D. 1. Reactor Building automatically Isolates and Recirculation Dampers CLOSE
2. SBGT system automatically starts
3. Components in CAM, PCP, and H2/O2 systems isolate

Proposed Answer: C.
1. Reactor Building automatically Isolates and Mixing Dampers OPEN
2. Standby gas treatment (SBGT) system automatically starts
3. Refueling Floor Fans Trip

Explanation (Optional):These radiation monitors will perform the following actions.

1. SBGT starts and opens inlet and exhaust dampers
2. 66AOV-100A / 66AOV-100B SUPPLY ISOLATION CLOSED
3. 66AOV-101A / 66AOV101B EXHAUST ISOLATION CLOSED
4. 66FN-5A, 5B, and 5C SUPPLY FANS 2 of 3 RUNNING
5. 66FN-12A or 6FN-12B BELOW 369' RUNNING

These radiation monitors do not isolate other components. In addition the normal ventilation lineup is in service for mixing the RB environment.

Technical Reference(s):AOP-15, ISOLATION VERIFICATION AND RECOVERY; REACTOR BUILDING VENTILATION AND COOLING SYSTEM* OP-51A and SDLP-o1B & 66A.

Proposed references to be provided to applicants during examination: None

Learning Objective: K1. Knowledge of the physical connections and/or cause- effect relationships between RADIATION MONITORING SYSTEM and the following: K1.10 Reactor building refuel floor: 3.4/ 3.6

Question Source: Bank # FitzPatrick Requalification Bank 2043
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7, 9_
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_2_	_2_
	Group #	_2_	_1_
	K/A #	290001 A1.01	
	Importance Rating	_3.1_	_3.1_

Proposed Question: 66 / 80

The unit is in a refueling outage with fuel moves in progress. The reactor building ventilation system is maintaining secondary containment at a negative 0.28 inches water differential pressure to atmosphere. The "A" standby gas treatment (SGT) train is operable and the "B" SGT train is tagged out of service for maintenance. Several minutes ago a fuel bundle was dropped over the core. The following indication are available in the control room.

Refueling Floor Exhaust
Radiation Monitor 17RM-456A ....1.2 x 10E 2 cpm

Refueling Floor Exhaust
Radiation Monitor 17RM-456B ....3.2 x 10E 5 cpm

Reactor Building Vent
Radiation Monitor 17RM-452A ....1.5 x 10E 2 cpm

Reactor Building Vent
Radiation Monitor 17RM-452B ....1.3 x 10E 2 cpm

Based on this information what is the expected configuration of secondary containment.

A. "A" SGT Train is ON, "B" SGT Train is OFF
Supply Isolation Valves 66AOV-100A & B CLOSED
Exhaust Isolation Valves 66AOV-101A & B CLOSED

B. "A" SGT Train is ON, "B" SGT Train is OFF
Supply Isolation Valve 66AOV-100A CLOSED
Exhaust Isolation Valve 66AOV-101A CLOSED

C. "A" & "B" SGT are OFF
Supply Isolation Valve 66AOV-100A CLOSED
Exhaust Isolation Valve 66AOV-101A CLOSED

D. "A" & "B" SGT are OFF
Supply Isolation Valve 66AOV-100B CLOSED
Exhaust Isolation Valve 66AOV-101B CLOSED

Proposed Answer: D. "A" & "B" SGT are OFF
Supply Isolation Vale 66AOV-100B CLOSED
Exhaust Isolation Valve 66AOV-101B CLOSED

Explanation (Optional): In this condition the 17RM-456B has detected the high radiation. The "A" has failed as is. Base on this information the "A" SGT system will not start and since

the “B” is out of service there will be no SGT running. The reactor building will have the “B” isolation valve close.

Technical Reference(s):LER 98001, ARP 09-75-1-15, OP-51A, OP-31pp 52

Proposed references to be provided to applicants during examination: None

Learning Objective: A1. Ability to predict and/or monitor changes in parameters associated with operating the SECONDARY CONTAINMENT controls including: (CFR: 41.5 / 45.5) A1.01 System lineups 3.1/3.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__2__	__2__
	K/A #	290003 A4.04	
	Importance Rating	_2.8_	_3.0_

Proposed Question: 77 / 81

A loss of coolant accident has occurred. Secondary containment has isolated on high radiation in the reactor building vent exhaust. Annunciator 09-75-1-20, "CONTROL ROOM SUPPLY RADIATION MONITOR INOP OR HI," has just alarmed. What operator actions are required.

A. None. The control room HVAC system will isolate on a high control room supply radiation.
B. None. The control room HVAC System will isolate on a loss of coolant accident signal.
C. Place the control room ventilation isolation and purge control switch in PURGE to purge the control room area of any airborn radioactivity.
D. Place the control room ventilation isolation and purge control switch in ISOLATE to isolate the control room HVAC system.

Proposed Answer: D. Place the control room ventilation isolation and purge control switch in isolate to isolate the control room HVAC system.

Explanation (Optional): The control room HVAC does not have an automatic isolation function. On high radiation the operator must place the system in isolation mode. The use of purge mode is to remove smoke and stale air. It will increase the airborn activity in the control room if used under these circumstances.

Technical Reference(s):SDLP 70 and OP-55B, ARP 09-75-1-20

Proposed references to be provided to applicants during examination: None

Learning Objective: A4. Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.04 Environmental conditions 2.8/ 3.0

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	300000 A3.02	
Importance Rating	_2.9_	_2.7_

Proposed Question: 78 / 82

The Plant is operating at 100% power. The "A" air compressor is operating in LEAD, the "B" air compressor is operating in LAG and the "C" air compressor is in standby. After you received alarm AIR COMPRESSOR DISCHARGE TEMPERATURE HI on the "A" compressor, you verify the "A" second stage discharge air temperature is at 355°F and rising. As the temperature rises how will the instrument air system respond to this event? Assume no operator action.

A. The "A" and "B" air compressor will trip on high second stage discharge air temperature and when the air header pressure drops to 100 psig the "C" air compressor will automatically start.

B. The "A" air compressor will continue to run to failure because there is no high temperature trip on the second stage discharge air temperature. The "C" air compressor will automatically start when the breaker for the "A" air compressor opens.

C. The "A" air compressor will trip on high second stage discharge air temperature and when the air header pressure drops to 90 psig the "C" air compressor will start and maintain the air header pressure.

D. The "A" air compressor will trip on high second stage discharge air temperature and when the air header pressure drops to approximately 110 psig the "B" air compressor will load and maintain the air header pressure.

Proposed Answer: D. The "A" air compressor will trip on high second stage discharge air temperature and when the air header pressure decreases to approximately 110 psig the "B" air compressor will load and maintain the air header pressure.

Explanation (Optional): As the discharge air temperature on the "A" compressor increases it will trip at 360F. When the compressor trips the "B" which is operating in LAG will automatically load at 110 psig and maintain header pressure. The "C" which is in standby mode will not start because header pressure will not decrease to 100 psig.

A. Both running compressors will not trip on a high discharge temperature. Each compressor has its own temperature switch39TS-113A-C.

B. The "A" air compressor will trip at a temperature of 360.

C. The "C" compressor starts at 100 psig not 90 and the "B" will load and maintain the header pressure so that the "C" should not automatically start.

Technical Reference(s): SDLP-39, ARP 9-6-2-17, OP-39.

Proposed references to be provided to applicants during examination: None

Learning Objective: A3. Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including: (CFR: 41.7 / 45.7) A3.02 Air temperature 2.9/2.7

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 ____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	400000 A2.01	
Importance Rating	_3.3_	_3.4_

Proposed Question: 79 / 83

Lowering turbine building closed loop cooling (TBCLC) pressure will result in the standby pump starting at _(1)_ psig. If the TBCLC pressure can not be restored then _(2)_ .

A. (1) 75
(2) emergency service water will automatically start at 40 psig and inject into the TBCLC header.

B. (1) 75
(2) manually scram the reactor.

C. (1) 85
(2) emergency service water will automatically start at 40 psig and inject into the TBCLC header.

D. (1) 85
(2) manually scram the reactor.

Proposed Answer: D. (1) 85
(2) manually scram the reactor.

Explanation (Optional): The standby pump automatically starts at TBCLC discharge header pressure of 85 psig and if the standby pump fails to start then manually scram the reactor in accordance with AOP-47.

Technical Reference(s): AOP-47 Loss of TBCLC

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: (CFR: 41.5 / 45.6) A2.01 Loss of CCW pump 3.3/ 3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:	55.41	__10_
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__3__	__3__
	K/A #	215001 K6.04	
	Importance Rating	_3.1_	_4.1_

Proposed Question: 80 / 84

During a full core LPRM calibration a Traversing In-Core Probe (TIP) detector becomes stuck in the core. The detector can not be moved in or out. If a loss of coolant accident were to occur with a valid containment isolation signal what operator action, if any must be taken to isolate the TIP system penetration?

A. No operator action is required. The TIP Shear valve automatically fires to cut the detector cable and seal the guide tube.
B. No operator action is required. The Guide tube ball valve automatically closes, to cut the detector cable and seal the guide tube.
C. Operator action is required. The TIP shear valve must be manually fired to cut the detector cable and seal the guide tube.
D. Operator action is required. The Guide tube ball valve must be manually closed to cut the detector cable and seal the guide tube.

Proposed Answer: C. Operator action is required. The TIP shear valve must be manually fired to cut the detector cable and seal the guide tube.

Explanation (Optional): The TIP ball valve does not have enough force to cut the detector cable. The TIP shear valve does not have an automatic function. Under these condition the TIP shear valve must be used because the detector is stuck and the penetration must be isolated.

Technical Reference(s): SDLP-07F, RAP-7.3.14

Proposed references to be provided to applicants during examination: None

Learning Objective: K6. Knowledge of the effect that a loss or malfunction of the following will have on the TRAVERSING IN-CORE PROBE: (CFR: 41.7 / 45.7)
K6.04 Primary containment isolation system: 3.1/3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__3__	__3__
K/A #	233000 A1.03	
Importance Rating	_3.3_	_3.6_

Proposed Question: 81 / 85

The plant is maintaining 100% power. Several hours ago a leak on the fuel pool cooling pump discharge line resulted in the fuel pool cooling pumps tripping on low skimmer surge tank level. Maintenance expects to return the fuel pool cooling system to service in the next 5 hours. What temperature indications are available, under these conditions, to monitor fuel pool water temperature?

A. Fuel pool temperatures on the RHR & HPCI TEMP 10TRS-131 at the 09-21 panel.
B. Observing local annunciator alarm lights at fuel pool filter demineralizer panel
C. Running the residual heat removal system in fuel pool cooling assist and monitoring RHR heat exchanger temperatures.
D. Installing a temporary temperature indication in the spent fuel pool.

Proposed Answer: D. Installing a temporary temperature indication in the spent fuel pool.

Explanation (Optional):
A. This chart recorder only has fuel pool HX inlet and outlet as well as pump suction temperatures. The water in this area is no longer communicating with the fuel pool.
B. This local panel does not have any temperature alarms on it.
C. To use RHR the plant must be in cold shutdown. There is not enough time to get the plant to a cold shutdown condition before fuel pool cooling is expected to be returned to service.
D. Procedure AOP-68 FUEL POOL COOLING AND CLEANUP TROUBLE* has the operator install a temperature indication in the fuel pool as need for this type of event.

Technical Reference(s): AOP-68 FUEL POOL COOLING AND CLEANUP TROUBLE

Proposed references to be provided to applicants during examination: None

Learning Objective: AI. Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: (CFR: 41.5 / 45.5) AI.03 Pool temperature 3.1/ 3.3

Question Source: New X

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:	55.41	__5__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	_____
Group #	__3__	_____
K/A #	234000 K1.04	
Importance Rating	_3.3_	_3.6_

Proposed Question: R86

Given the following conditions:

The Mode Switch is in REFUEL.
Core Reload is in progress.
All control rods are fully inserted.
There are no other Rod Blocks in effect.
The only hoist in use is the Main Grapple Hoist.

And the following actions occur in the order provided.

1. One control Rod is selected.
2. A fuel assembly is grappled in the Fuel Pool and raised to Full-Up.
3. The Refuel Platform is driven over the core.
4. The fuel assembly is lowered into its assigned location.
5. The NCO1 attempts to withdraw the selected control rod.

The Rod Block first occurred when ...

A. The loaded Main Grapple Hoist reaches Full-Up in the Fuel Pool.
B. The Refuel Platform is driven over the core.
C. The Main Grapple Hoist starts to lower the fuel assembly.
D. The operator attempts to withdraw the selected control rod.

Proposed Answer: B. The Refuel Platform is driven over the core.

Explanation (Optional):
A. The rod block occurs when a control rod is selected and the bridge is over the core with a hoist loaded or the main grapple not full up.
C. This will be a rod block however, not the first one.
D. In Refuel you are allow to withdrawal one control rod, however with the bridge over the core and mast down there would be a rod block.

Technical Reference(s): SDLP-08B

Proposed references to be provided to applicants during examination: None

Learning Objective: K1. Knowledge of the physical connections and/or cause- effect relationships between FUEL HANDLING EQUIPMENT and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.04 †Reactor manual control system: 3.3 / 3.6

Question Source: Bank # INPO 8767

At which point in the following Core Alteration scenario would activation of the ROD OUT BLOCK annunciator first occur? (1C05B, A-6)

Scenario:

-The Mode Switch is in REFUEL.
-All control rods are fully inserted.
-One control Rod is selected.
-There are no other Rod Blocks in effect.
-The only hoist in use is the Main Grapple Hoist.
-A fuel assembly is grappled and raised to Full-Up in the Fuel Pool.
-The Refuel Platform is driven over the core.
-The fuel assembly is lowered into its assigned location.
- The 1C05 operator attempts to withdraw the selected control rod.

Rod Block first occurs when ...

the Refuel Platform is driven over the core.
the loaded Main Grapple Hoist reaches Full-Up in the Fuel Pool.
the Main Grapple Hoist starts to lower the fuel assembly.

the 1C05 operator attempts to withdraw the selected control rod.
Reference: ..234000.K4.01

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __2__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	_____
Group #	__3__	_____
K/A #	288000 K1.02	
Importance Rating	_3.4_	_3.4_

Proposed Question: R87

Secondary containment (reactor building to atmosphere) differential pressure is being maintained at a negative 0.28 inches water with the following line up.

Main Supply Fan	FN-5A ....OPERATING
Main Supply Fan	FN-5B ....OPERATING
Main Supply Fan	FN-5C ....STANDBY
Below Refuel Floor Exhaust Fan	FN-12A ....OPERATING
Below Refuel Floor Exhaust Fan	FN-12B ....STANDBY
Refuel Floor Exhaust Fan	FN-13A ....OPERATING
Refuel Floor Exhaust FAn	FN-13B ....STANDBY

If the Refuel Floor Exhaust fan FN-13A tripped on overcurrent what effect would this have on secondary containment?

A. The main supply fans will automatically trip to maintain a negative secondary containment differential pressure.
B. The refuel floor exhaust fan FN-13B would automatically start to maintain a negative secondary containment differential pressure.
C. The refuel floor exhaust fan FN-13B must be manually started to maintain a negative secondary containment differential pressure.
D. The below refuel floor exhaust fan FN-12B would automatically start to maintain a negative secondary containment differential pressure.

Proposed Answer: B. The refuel floor exhaust fan FN-13B would automatically start to maintain a negative secondary containment differential pressure.

Explanation (Optional):
A. The supply fans will automatically trip if both below refuel floor exhaust fans are off, not the above refuel floor fans..
C. The start of FN-13B is an automatic start not a manual start.
D. This fan will automatically start if the other below refuel floor exhaust fan trips not the refuel floor exhaust fan.

Technical Reference(s):SDLP-66A

Proposed references to be provided to applicants during examination: None

Learning Objective: K1. Knowledge of the physical connections and/or cause- effect relationships between PLANT VENTILATION SYSTEMS and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.02 Secondary containment 3.4/ 3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__9__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__3__	__3__
Group #	_____	_____
K/A #	2.1.10	
Importance Rating	_2.7_	_3.9_

Proposed Question: 85 / 88

The plant has been at a steady state power for the last 7 days during a feedwater flow calibration. The calibration is complete and the final adjustments have been made to the feedwater flow instrument loop. The first 3D Monicore output after these adjustments were made has the following information:

POWER MWT 2580
POWER MWE 860
FLOW MLB/HR 75.453
MFLCPR 0.938 37-24
MFLPD 0.946 41-24-6
PCRAT 1.004 15-32-4
PR (PSIa) 1057
LOAD LINE 101.7%

Based on this information what actions must the operator take?

A. Reduce reactor pressure to provide more margin to the high pressure reactor scram.
B. Thermal power must be reduced to 2536 MWt or less within 15 minutes.
C. Contact reactor engineering within 15 minutes to decrease PCRAT less than 1.0.
D. The reactor must be scrammed.

Proposed Answer: B. Thermal power must be decrease to 2536 MWt or less within 15 minutes.

Explanation (Optional):
A. MFLCPR is less than 1.0 as required by TS
B. Correct - RAP 7.3.16 requires the plant to reduce power to less than 2536 MWt within 15 minutes.
C. There is no need to reduce PCRAT to less than 1.0, because the plant has been at this power level for 7 days. The envelop must be updated.
D. There is no reason to scram the reactor.

Technical Reference(s): RAP 7.3.16

Proposed references to be provided to applicants during examination: None

Learning Objective: Knowledge of conditions and limitations in the facility license. RO 2.7 / SRO 3.9

LO NET 238.3, 1.03.j Rated Thermal Power

Question Source: Bank # FitzPatrick Requal 33301036COLRS01 Rev.1

The plant is at full power when it is discovered that actual thermal power is 2580 MWt.

Select the following required action:

A. Scram the reactor.

B. Reduce power to within 2536 MWt within 15 minutes.

C. Commence an orderly shutdown by inserting the RSCS groups per RAP-7.3.16.

D. Quickly insert the cram rods per RAP-7.3.16 and notify the Nuclear Engineer.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _5, 10
55.43 _____

Comments:

Examination Outline Cross-reference: Level		RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	2.1.19	
	Importance Rating	_3.0_	_____

Proposed Question: R89

A valid Group II isolation signal has occurred. All Group II valves closed except one drywell floor drain sump isolation valve. Based on this information what color will be indicated on the EPIC computer for Group II isolation status.

A. RED
B. GREEN
C. GRAY
D. MAGENTA

Proposed Answer: B. GREEN

Explanation (Optional):
A. If one line is not isolated. Both valves in one line OPEN
B. All lines are isolated. At least one valve in each line is CLOSED.
C. If no isolation signal is present
D. EPIC has lost data or sensors for this parameter

Technical Reference(s): SDLP-66A

Proposed references to be provided to applicants during examination: None

Learning Objective: Ability to use plant computer to obtain and evaluate parametric information on system or component status. RO 3.0/ SRO 3.0

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	3	
Group #		
K/A #	2.1.22	
Importance Rating	2.8	

Proposed Question: R90

Preparations are being made to startup the plant. The following conditions exist:

Reactor Mode Switch ...Shutdown
Reactor Pressure ....125 psig
Reactor Head Bolts ......Fully tensioned
Control rods ....All rods are IN

The reactor is in the:

A. Refueling Mode.
B. Hot Shutdown Mode.
C. Cold Shutdown Mode.
D. Startup/Hot Standby Mode.

Proposed Answer: B. Hot Shutdown Mode

Explanation (Optional): The mode switch is in the Shutdown position and reactor water temperature is greater than 212F. Therefore hot shutdown.

Technical Reference(s): TS Definition section

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.1.22 Ability to determine Mode of Operation
NET-238.3, EO 1.03.g Mode of Operation

Question Source: Bank # INPO 7976

Preparations are presently being made to startup the Unit One reactor. The following conditions exist:
Reactor Mode Switch: Shutdown
Reactor Pressure: 125 psig
All reactor vessel head closure bolts are fully tensioned All rods are IN.

The reactor is in:

Mode 3
Mode 2
Mode 4
Mode 5

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	_____	
	55.43	_____	
	55.45	13	NET-238.3, EO 1.03.g Mode of Operation

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	3	
Group #		
K/A #	2.2.2.	
Importance Rating	4	

Proposed Question: R91

A plant startup is in progress and reactor water recirculation pump speed is 30%. The average power range monitor readings are listed below:

Channel A 2.3%
Channel B 4.0%
Channel C 5.0%
Channel D 4.7%
Channel E 2.4%
Channel F 5.1%

If the reactor mode switch is placed in RUN, which one of the following describes the ability to raise power with control rods and/or recirc. flow?

A. Power can be raised using control rods or reactor water recirculation flow.
B. Power can be raised using control rods only.
C. Power can be raised using reactor water recirculation only.
D. Power cannot be raised.

Proposed Answer: D. Power cannot be raised.

Explanation (Optional): Under these conditions if the mode switch is placed in RUN the APRMs will give a rod block and feedwater flow is less than 20% so that the RWR pump speed can not be raised.

Technical Reference(s): None

Proposed references to be provided to applicants during examination: None

Learning Objective: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels. RO 4.0 / SRO 3.5

Question Source: Bank # INPO 7674

A plant startup is in progress and Recirc. pump speed is 30%. APRM readings are as listed below:

Channel A 2.3%
Channel B 4.0%
Channel C 5.0%
Channel D 4.7%
Channel E 2.4%
Channel F 5.1%

If the Reactor Mode Switch is placed in RUN, which one of the following describes the ability to raise power with control rods and/or recirc. flow?

A. Power can be raised using control rods or reactor water recirculation flow.
B. Power can be raised using control rods only.
C. Power can be raised using reactor water recirculation only.
D. Power cannot be raised.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__6__
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	2.2.23	
	Importance Rating	_2.6_	_____

Proposed Question: R92

The standby liquid control pump is being removed from service. In which of the following logs can an individual conclusively determine the status of the redundant plant equipment?

A. Night Order Book
B. PTR Log
C. LCO Binder
D. Equipment Status Log

Proposed Answer: C. LCO Binder

Explanation (Optional):
A. This is a status of selected components, not all components.
B. This is a status of plant tagging, it does not / may not contain information on inoperable system from engineering, only tagging.
C. This log must have all LCO on all equipment that LCO are required to be tracked. SLC is a TS system and must be tracked.
D. This will not show all equipment LCOs

Technical Reference(s): ODSO-34 TECH SPEC LCO AND MAINTENANCE RULE UNAVAILABILITY TRACKING pp 12

Proposed references to be provided to applicants during examination: None

Learning Objective: Ability to track limiting conditions for operations. RO 2.6 / SRO 3.8

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __10_
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__3__	_____
Group #	_____	_____
K/A #	2.2.12	
Importance Rating	_3.0_	_____

Proposed Question: R93

Surveillance test ST-24J, "RCIC Flow Rate and Inservice Test (IST)," is being performed. The following conditions are present.

Torus Temperature ....87°F
Torus Level ....13.9 Feet
RCIC Flow ....400 gpm
RCIC discharge Pressure ....1500 psig
RCIC Turbine speed ....2525 rpm
RHR Loop "A" ....IN Torus Cooling

Based on these conditions, what operator actions are required?

A. The test must be terminated and torus temperature reduced to less than 85°F within 24 hours.
B. The test must be terminated and execute EOP-4 to reduce torus water level.
C. Raise turbine speed by throttling 13MOV-30, test valve to CST.
D. Reduce RCIC discharge pressure by throttling 13MOV-30, test valve to CST.

Proposed Answer: D. Reduce RCIC discharge pressure by throttling 13MOV-30, test valve to CST.

Explanation (Optional):
A. Torus temperature must be less than 105 F and reduced to less than 95F within 24 hours.
B. Torus water level is in the normal band and EOP-4 does not have to be entered.
C. Turbine speed is above 2200 rpm and rasing turbine speed would further rase pressure.
D. The pressure is above the design pressure of 1320psig

Technical Reference(s): ST-24J, "RCIC Flow Rate and Inservice Test (IST)

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.2.12 Knowledge of surveillance procedures.

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 __10_	
	55.43 ______	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	2.3.4	
	Importance Rating	_2.5_	_____

Proposed Question: R94

An operator will receive 700 mrem over the next 3 weeks performing a special task during the refueling outage. The operator has an accumulated TEDE dose this year of 2900 mrem. What action is required for the operator to perform this task?

A. No action is required.
B. Only the Radiation Protection Manager must approve the operator exceeding the administrative TEDE dose guideline.
C. Only the Operations and Radiation Protection Manager must approve the operator exceeding the administrative TEDE dose guideline.
D. The Radiation Protection Manager, Plant Manager and Site Executive Officer must approve the operator exceeding the administrative TEDE dose guideline.

Proposed Answer: A. No action is required.

Explanation (Optional): The site TEDE dose guideline is 4000 mrem / year. For this example the dose projected (3600 mrem) is below the TEDE dose guideline. Approval to exceed this is required by the Radiation Protection Manager, Plant Manager and Site Executive Officer. AP-07.05 pp. 22

Technical Reference(s):AP-07.05 pp. 22

Proposed references to be provided to applicants during examination: None

Learning Objective: Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. RO 2.5 / SRO 3.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _10__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	3	
Group #		
K/A #	2.3.10	
Importance Rating	2.9	

Proposed Question: R 95

In accordance with Technical Specifications which of the following is the minimum proper control(s) for any accessible area with a dose rate of 1100 mrem / hr?

A. post the door as a radiation area.
B. post the door as a high radiation area.
C. lock the door and post the door as a high radiation area.
D. lock the door and post the door as a very high radiation area.

Proposed Answer: C. lock the door and post the door as a high radiation area.

Explanation (Optional): Dose rates in the area of the moisture separator are 1100 mrem/hr. The door should be locked and controlled as a high rad area.

Technical Reference(s): AP-07.06 HIGH RADIATION AREA CONTROL, TS 6.11

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.3.11, Ability to perform procedure to reduce excessive levels of radiation and guard against personnel exposure.

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 _____
55.43 __4__

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__3__	__3__
	Group #	_____	_____
	K/A #	2.3.11	
	Importance Rating	_2.7_	_3.2_

Proposed Question: 95 / 96

A main steam line break outside of containment has occurred. The main steam line isolation valves have successfully isolated the break and the reactor scrammed. What actions can be taken to reduce radiation leakage to the environment in accordance with AOP-40 MAIN STEAM LINE BREAK.

A. Dispatch a team to verify reactor building integrity and start the main steam leakage collection system from the control room.
B. Dispatch a team to verify turbine building integrity and start the main steam leakage collection system from the relay room.
C. Dispatch a team to verify turbine building integrity and start the main steam leakage collection system from the remote shutdown panels.
D. Perform an emergency reactor depressurization.

Proposed Answer: B. Dispatch a team to verify turbine building integrity and start the main steam leakage collection system from the relay room.

Explanation (Optional): A. The MSLC system is manually operated from the relay room and the procedure directs teams to be dispatched to the turbine building. An RPV emergency depressurization is not required because there is no primary system discharging into secondary containment.

Technical Reference(s):AOP-40 MAIN STEAM LINE BREAK

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.3.11, Ability to control radiation releases.

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__3__	__3__
Group #	_____	_____
K/A #	2.4.1	
Importance Rating	_4.3_	_4.6_

Proposed Question: 96 / 97

The plant was operating at 100 % power when an event occurs. The following plant conditions are present:

APRM Reactor Power	15%	
Reactor Level	200 inches	
Reactor Pressure	1145 psig	
Drywell Temperature	135°F	
Drywell Pressure	2.9 psig	
Torus Water Temperature	95°F	
Torus Pressure....................................	1.0 psig	
Torus Level	13.9 Feet	
Reactor Building to outside dP	- 0.29 inches water	
Reactor Building Temperatures	RWCU heat exchanger room	105°F
	RWCU "A" Pump Room	110°F
	RWCU "B" Pump Room	110°F
Reactor Building Radiation Levels	RWCU Heat Exchanger Room	30 mr/hr
	RWCU Pump Area	20 mr/hr

Based on these conditions what emergency operating procedures should be entered/used and what operator actions have been taken in the first minute?

A. EOP-2, RPV Control; EOP-3 Failure to Scram; and EOP-4, Primary Containment Control have been entered and drywell sprays have been initiated.

B. EOP-2, RPV Control; EOP-3 Failure to Scram; and EOP-4, Primary Containment Control have been entered and a manual scram initiated and mode switch placed in shutdown.

C. EOP-3 Failure to Scram; EOP-4, Primary Containment Control; and EOP-5 Secondary Containment Control have been entered and the mode switch has been taken to shutdown.

D. EOP-3 Failure to Scram; EOP-4, Primary Containment Control; and EOP-4a, Primary Containment Gas Control have been entered and SRMs & IRMs are driving into the core.

Proposed Answer: B. EOP-2, RPV Control; EOP-3 Failure to Scram; and EOP-4, Primary Containment Control have been entered and a manual scram initiated and mode switch placed in shutdown.

Explanation (Optional):
A. There are no conditions that would require drywell sprays to be initiated.
C. There are no entry condition for EOP-5.
D. There is adequate core cooling, therefore, there is no entry conditions for EOP-4a.

Technical Reference(s):FSAR Section 14, EOPs

Proposed references to be provided to applicants during examination: EOPs w/o entry conditions.

Learning Objective: 2.4.1 Knowledge of EOP entry conditions and immediate action steps.

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __10__
55.43 __5___

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__3__	__3__
Group #	_____	_____
K/A #	2.4.18	
Importance Rating	_2.7_	_3.6_

Proposed Question: 97 / 98

In EOP-4, "Primary Containment Control," the primary containment pressure section (PC/P) requires that torus pressure exceeds 15 psig before drywell sprays can be initiated. Why must torus pressure exceed 15 psig prior to initiating drywell sprays?

A. This will prevent chugging in the drywell vent downcomers which could result in structural damage to the downcomers.
B. This will ensure a large enough differential pressure to open the torus to drywell vacuum breakers and reduce the torus pressure.
C. This will remove radioactivity from the noncondensibles, which must be performed prior to venting the containment.
D. The higher torus pressure will reduce the upward water force on the drywell vent header as the water in the downcomer is blown out of the downcomer.

Proposed Answer: A. This will prevent chugging in the drywell vent downcomers which could result in structural damage to the downcomers.

Explanation (Optional):
B. The torus to drywell vacuum breakers only requires a 0.5 psid to open. They do not need the high torus pressure 15 psig.
C. It is recommended that the venting is performed through the torus to scrub. It is not a requirement that this be performed prior to venting the containment.
D. The water level in the downcomer effects the upward forces on the drywell vent header. This is only a concern during the initial break because after the torus reaches 15 psig the RPV is in a lower energy state and will not be able to produce the forces that were produce prior to the primary system break.

Technical Reference(s): MIT-301.11E, pp 18

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.4.18 Knowledge of the specific basis for EOPs

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:	55.41	__10_
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__3__	_____
	Group #	_____	_____
	K/A #	2.4.24	
	Importance Rating	_3.3_	_____

Proposed Question: R99

A loss of coolant accident (LOCA) has occurred coincident with a loss of offsite power. The control room supervisor directs supplying emergency service water (ESW) to the drywell coolers per AOP-11, "Loss of Reactor Building Closed Loop Cooling (RBCLC)." Which one of the following describes why this order is not appropriate under these conditions.

A. Suppling ESW to the drywell coolers would require shutting down one pair of emergency diesel generators.
B. ESW has already automatically align to supply the drywell coolers due to the loss of RBCLC.
C. Suppling ESW to the drywell coolers could cause failure of RBCLC piping due to water hammer.
D. Suppling ESW to the drywell coolers may cause a rapid reduction of drywell pressure causing reactor building to torus vacuum breakers to open.

Proposed Answer: C. Suppling ESW to the drywell coolers could cause failure of RBCLC piping due to water hammer.

Explanation (Optional): AOP-11, cautions that suppling ESW or reinjecting RBCLC to drywell after LOCA is prohibited because this could cause failure of the RBCLC piping inside drywell and jeopardize primary containment. In addition, ESW to the drywell coolers is a manually alignment not automatic.

A. This does not require shutdown of a DG pair.
B. This will not cause a rapid reduction of DW pressure and AOP-11 does not caution against this from occurring.

Technical Reference(s): AOP-11, Loss of Reactor Building Closed Loop Cooling

Proposed references to be provided to applicants during examination: None

Learning Objective: Knowledge of loss of cooling water procedures. RO 3.3 / SRO 3.7

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:	55.41	__10_
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__3__	__3__
	Group #	_____	_____
	K/A #	2.4.32	
	Importance Rating	_3.3_	_3.5_

Proposed Question: 99 / 100

The reactor is at 75% power on a high rod line and is being rased to 100% power with recirculation flow. Torus cooling is inservice and personnel are being staged for the performance of ST-4N, "HPCI Quick-Start, Inservice and Transient Monitoring Test (IST)." Once the personnel are staged power ascension will be stopped for HPCI testing. A moment ago, the operators reported that the following annunciators went dark without re-flash.

(1) 09-4-1-26 CORE SPRAY OR RHR PUMP RUNNING
(2) 09-5-2-2 ROD WITHDRAWAL BLOCK

The APRM hi lights are still flashing and the RHR system is still operating in torus cooling. EPIC computer alarm LOSS OF PWR INTERPOSING RLY SYS has occurred. What has occurred to the annunciators on the 09-3 through the 09-8 panels and what actions must be taken?

A. The AC annunciator power has been lost and automatically transferred to DC power, the annunciators must be reset from panels IR-1 & IR-2 in the relay room. Power ascension and testing may continue while the annunciators are reset.
B. The AC annunciator power has been lost and must be manually transferred to DC power at panels IR-1 & IR-2 in the relay room. The power ascension and all testing will be stopped during the manual transfer to DC power.
C. The AC and DC annunciator power has been lost. Power ascension and all testing will be stopped.
D. The AC and DC annunciator power has been lost. Power ascension will be stopped; however, HPCI testing should be performed while maintenance restores the annunciators.

Proposed Answer: C. The AC and DC annunciator power has been lost. Power ascension and all testing will be stopped.

Explanation (Optional):
A. If AC power is lost and automatically transfers to DC power the annunciators do not have to be reset.
B. If AC power fails the transfer to DC backup power is automatic.
D. If annunciators are lost all power ascension and test must be stopped in accordance with AOP-65.

Technical Reference(s): AOP-65

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.4.32 Knowledge of operator response to loss of all annunciators.

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __10_
55.43 __5__

Comments:

James A. FitzPatrick

November 2001

SRO Written Exam

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295003 AA1.03	
Importance Rating	_4.4_	_4.4_

Proposed Question: 1 / 16

A station blackout has occurred. What action is necessary to ensure the plant can be safely shutdown.

A. The high pressure coolant injection (HPCI) system should be used preferentially over the reactor core isolation cooling (RCIC) system for RPV make up because of the higher flow rate.

B. The RCIC system should be used preferentially over HPCI for RPV makeup because RCIC is less likely to cycle on and off due to RPV water level.

C. Align the RCIC suction to the Torus since there is no power available for make up water to the CST.

D. Verify that all fire doors in the HPCI and RCIC room are closed so that a steam line break will not result in the loss redundant systems simultaneously.

Proposed Answer: B. The reactor core isolation cooling (RCIC) system should be used preferentially over HPCI for RPV makeup because RCIC is less likely to cycle on and off due to RPV water level.

Explanation (Optional):
A. HPCI has a larger capacity and will cycle more than RCIC. This will result in less battery operational time.
C. RCIC low CST level should be bypassed because operation with torus water at elevated temperatures will reduce the RCIC reliability.
D. The doors are required to be opened to reduce local area temperatures because of Steam leakage from HPCI and RCIC turbine shaft seals that could be encountered. The leakage could be caused by a lack of gland seal suction or higher than normal turbine exhaust pressures.

Technical Reference(s): AOP-49, Station Blackout

Proposed references to be provided to applicants during examination: None

Learning Objective: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.7 / 45.6) AA1.03 Systems necessary to assure safe plant shutdown 4.4*/ 4.4

Question Source:
Bank # ________
Modified Bank # _____ (Note changes or attach parent)
New __X__

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1___
Group #	__1__	__1___
K/A #	295006 AA2.02	
Importance Rating	_4.3_	_4.4_

Proposed Question: 2 / 1

The plant has just experienced an ATWS in which several control rods had their scram inlet and outlet valves fail to open and did not fully insert. The control room supervisor orders that these control rods be manually inserted using RMCS. Based on the following indication on the full core display, which one of the following control rods must be inserted?

A. Control rod 22-27 has the BLUE light ON
B. Control rod 22-35 has a position indication of 00 on the four rod display
C. Control rod 18-31 has the RED light ON
D. Control rod 26-39 has the GREEN light ON

Proposed Answer: C. Control rod 18-31 has the RED light ON

Explanation (Optional): A. BLUE light lit means that both scram valves opened and full in by stem
B. Reactor will be shutdown under all conditions if rods at 00 or 02, 00 is full in
D. GREEN light ON means rod is full inserted
C. RED indication is full withdrawn.

Technical Reference(s): SDLP-03F, Reactor Manual Control; EP-3, Backup Manual Control Rod Insertion

Proposed references to be provided to applicants during examination: None

Learning Objective: Ability to determine and/or interpret the following as they apply to SCRAM: (CFR: 41.10 / 43.5 / 45.13) AA2.02 Control rod position 4.3*/ 4.4*

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content:
55.41 __10_
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1___
Group #	__1__	__1___
K/A #	295007 2.3.4	
Importance Rating	_2.5_	_3.1_

Proposed Question: 3 / 2

A large break loss of coolant accident (LOCA) has occurred. All low pressure ECCS systems started except the "A" & "C" RHR pumps. that were out of service with their discharge valves closed for planned maintenance. Even though adequate core cooling is met, the shift manager determined that the "A" & "C" RHR pumps need to be returned to service. Radiation protection has determined that the dose rate by the pump discharge valves is 60 Rem / hr. To meet the emergency exposure guideline the maximum time an individual has to open the discharge valves is __(a)__ minutes and his emergency exposure must be approved by the __(b)__ .

A. (a) 10
(b) Emergency Director

B. (a) 10
(b) TSC Manager

C. (a) 20
(b) Emergency Director

D. (a) 20
(b) TSC Manager

Proposed Answer: A (a) 10
(b) Emergency Director

Explanation (Optional): EAP-15, Emergency Radiation Exposure Criteria and Control limits the maximum dose to protect equipment to 10 Rem and this exposure must be approved by the emergency director.

Technical Reference(s): EAP-15, Emergency Radiation Exposure Criteria and Control

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.3.4 Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10) RO 2.5 / SRO 3.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X___

10 CFR Part 55 Content:	55.41	_____
	55.43	__4___

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295009 AA2.03	
Importance Rating	_2.9_	_2.9_

Proposed Question: 4 / 3

The reactor is critical at 160 psig. The feedwater level control system has been removed from service. Reactor water level has been stable and is being controlled via the "A" CRD pump and reactor water clean up (RWCU) blowdown at a blowdown rate of 60 gpm. Several minutes ago the "A" CRD pump tripped. There are no CRD accumulators alarms are present. The operator must __(1)__ RWCU blowdown rate to prevent __(2)__ .

A. (1) Reduce
(2) a HPCI / RCIC injection signal

B. (1) Reduce
(2) a reactor scram

C. (1) Raise
(2) a HPCI / RCIC trip signal

D. (1) Raise
(2) a reactor scram

Proposed Answer: B. (1) Reduce
(2) a reactor scram

Explanation (Optional):

A. This condition will never occur. The water level reduction will be terminated when level reaches 177 and RWCU isolates. This is above the 126.5 level for RCIC and HPCI start.

B. Correct - if the operator does not reduce the blowdown then a reactor scram / GP II / RWCU isolation will occur.

C. This condition will not occur. If the blowdown is raised the level will drop, not rise.

D. If the blowdown is increased the low level scram / isolations will occur.

Technical Reference(s):SDLP-12, RWCU, OP-23, OP-65, AOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: AA2. Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) AA2.03 Reactor water cleanup blowdown rate 2.9 / 2.9

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__5___
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295010 AA1.01	
Importance Rating	_3.4_	_3.5_

Proposed Question: 5 / 4

The plant is operating at 100% power when an operator mistakenly isolates the reactor building closed loop cooling (RBCLC) to the "A" Drywell Cooler. The operator notifies the control room of the mistake and states that he can not reestablish any flow to the "A" Drywell Cooler. What effect does this have on the containment and what operator actions would have to be taken.

A. Drywell temperature and pressure will remain steady because the redundant "B" drywell cooler is in service. No operator action will be required.

B. Drywell temperature and pressure will rise. Operator action will be required to maintain drywell pressure below 2.7 psig.

C. Drywell temperature and pressure will remain steady due to the stored heat capacity in the cooling water of the "A" drywell cooler. No operator action is required.

D. Drywell temperature and pressure will rise. Operator action is required to manually start the fourth fan on the "B" drywell cooler which will then be able to maintain drywell temperature below 135°F.

Proposed Answer: B. Drywell temperature and pressure will rise. Operator action will be required to maintain drywell pressure below 2.7 psig.

Explanation (Optional):

A. A loss of 50% cooling in the drywell at 100% power will result in a rapid heat up and pressure rise (minutes).

B. A loss of 50% cooling in the drywell at 100% power will result in a rapid heat up and pressure rise (minutes). The operator will enter EOP-4 on high drywell temperature and be required to maintain drywell pressure below 2.7 psig.

C. There is not sufficient heat capacity to maintain drywell temperature less than 135 by starting the forth fan on the "B" drywell cooler.

D. Starting the forth fan on the "B" cooler will not be able to maintain temperature below 135.

Technical Reference(s):EOP-4, OP-53, "DRYWELL VENTILATION AND COOLING."

Proposed references to be provided to applicants during examination: None

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.7 / 45.6) AA1.01 Drywell ventilation/cooling 3.4 / 3.5

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	_____	__1__
Group #	_____	__1__
K/A #	295013 AK3.02	
Importance Rating	_____	_3.8_

Proposed Question: S6

The plant is operating normally at 100% power. The High Pressure Coolant Injection (HPCI) system has been manually initiated to perform ST-4N, "HPCI Quick-Start, In service, and Transient Monitoring Test (IST)." The suppression pool water temperature is rising. TORUS BULK TEMP HI OR RTD FAILURE annunciator has alarmed. The operator observes the suppression pool water temperature to be 112 °. Which one of the following statements describes the next required action to be taken and the reason for the action.

A. Secure HPCI testing and operate all available suppression pool cooling to ensure that under the worst case accident conditions, sufficient net positive suction head would be available to core spray and the residual heat removal pumps.

B. Initiate an orderly power reduction to ensure that the suppression pool is maintained within the heat capacity temperature limit.

C. Insert a manual scram to ensure that if a reactor blowdown were to occur, stable steam condensation will occur during the blowdown.

D. Start water transfer from the suppression pool to condenser hotwell and establish makeup to the suppression pool from the CST's to increase the heat capacity of the suppression pool.

ANSWER: C. Insert a manual scram to ensure that stable steam condensation will occur during a reactor blow down.

Explanation (Optional): Technical Specification require a manual scram if suppression pool temperature is greater than 110 °F. This will ensure that the suppression pool will have stable steam condensation during the blowdown.

Technical Reference(s): ARP09-3-1-14, Technical Specification basis 3.7

Proposed references to be provided to applicants during examination: EOP-4, W/O entry conditions

Learning Objective: AK3. Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.5 / 45.6) AK3.02 Limiting heat additions 3.6 / 3.8

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _X_
Comprehension or Analysis ___

The EOP-4, Primary Containment requires a manual scram as well as TS, TS is memory an EOP is Comprehension.

10 CFR Part 55 Content:	55.41	__5___
	55.43	_(b)(2)_

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	_____	__1_
Group #	_____	__1_
K/A #	295014 AK2.11	
Importance Rating	_____	_3.7_

Proposed Question: S7

The reactor is operating at 100% power when the following annunciator alarms, RWR MG A DRV MTR BKR TRIP. You notice reactor power is dropping, the "A" reactor recirculation pump speed has dropped to zero and feedwater flow and steam flow are dropping. The operator has received several periodic LPRM upscales and downscale alarms on the full core display and 09-5 panel annunciators. Based on this information what actions are required and why must they be taken?

A. Reduce the "B" recirculation pump speed to 90% to minimize excessive jet pump differential pressures.
B. Start the "A" recirculation pump to provide increased margin to thermal-hydraulic instability.
C. A manual reactor scram is required because of thermal-hydraulic instability.
D. Determine if the MCPR Safety Limit has been exceeded and manually scram the reactor if the safety limit has been exceeded.

Proposed Answer: C. A manual reactor scram is required because thermal-hydraulic instability is occurring.

Explanation (Optional): The operating condition after a recirculation pump trip will place the unit near the exclusion region. AOP-8, "Loss of Coolant Flow," requires the operators to monitor for stability and if indications for thermal-hydraulic instability. is identified to manually scram the reactor. The periodic LPRM upscale and downscale alarms indicate that the core is unstable A manual scram is required.

A. Running pump must be less than 80%
B. Core instabilities are occurring must manually scram.
C. Indication of instability and manual scram is required
D. If a safety limit is exceeded the reactor must be shutdown. There is no requirement to scram. TS 6.7

Technical Reference(s): AOP-8 Loss of Coolant Flow, AOP-32Unexplained Reactivity Change

Proposed references to be provided to applicants during examination: None

Learning Objective: AK2. Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: (CFR: 41.7 / 45.8) AK2.11 Recirculation flow control 3.6 / 3.7

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__7___
	55.43	_(b)(5)_

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295015 AK1.01	
Importance Rating	_3.6_	_3.9_

Proposed Question: 8 / 7

The plant has just experienced a scram. All control rods fully inserted with the exception of control rod 26-27, which inserted to position 24. Subsequent attempts to insert the control rod were not successful. The control room supervisor orders a cool down in accordance with the EOPs. What actions are required, prior to the cool down to ensure that the reactor will remain shutdown during the cool down.

A. Control rod 26-27 must be fully inserted prior to beginning the cool down.
B. Standby liquid control must be injected prior to beginning the cool down.
C. Reactor engineering must perform calculations to prove the reactor will remain shutdown prior to beginning the cool down.
D. No actions are necessary, the reactor will remain shutdown during the cool down.

Proposed Answer: D. No actions are necessary, the reactor will remain shutdown during the cool down.

Explanation (Optional): shutdown margin requires that the reactor will remain shutdown under all conditions without boron injection with one control rod fully withdrawn (or any other position), provided all other control rods are inserted to or beyond position 02.

Technical Reference(s): EP-6, Backup control rod insertion

Proposed references to be provided to applicants during examination: None

Learning Objective: AK1. Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: (CFR: 41.8 to 41.10)
AK1.01 Shutdown margin 3.6* / 3.9*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __x___
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __8___
55.43 ______

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__1__
Group #	____	__1__
K/A #	295016 AK2.01	
Importance Rating	_____	_4.5_

Proposed Question: S9

The plant has just experienced a spurious fast closure of the "A" inboard and "D" outboard main steam line isolation valves; a reactor scram; and isolation of the high pressure coolant injection system. An operator has reported that there is fire and heavy black smoke in the cable spreading room. What actions must be taken, in accordance with station procedures?

A. Evacuate the control room and station operators at the auxiliary shutdown panels and place isolation switches in REMOTE to prevent spurious operation of equipment.
B. Evacuate the control room and station operators at the auxiliary shutdown panels and place isolation switches in LOCAL to prevent spurious operation of equipment.
C. Do NOT evacuate the control room, place the keylock switches on the 09-3 panel for 10MOV-25A(B) and 10MOV-27A(B) to BYPASS to prevent inadvertent actuation of these valves.
D. Do NOT evacuate the control room, place the keylock switches on the 09-3 panel for 10MOV-25A(B) and 10MOV-27A(B) to NORMAL to prevent inadvertent actuation of these valves.

Proposed Answer: B. Evacuate the control room and station operators at the auxiliary shutdown panel and alternate shutdown panel and place isolation switches in LOCAL to prevent spurious operation of equipment.

Explanation (Optional): AOP-43 requires that the control room be evacuated if (1) a significant fire exists in the cable spreading room (2) verbal report of a fire (3) unexplained loss of equipment. All three conditions are met. The isolation switches must be placed in LOCAL. This will isolate and remove all the automatic actions of the individual components. If the switches are in the remote positions then control is via the control room and not the remote shutdown panel.

Technical Reference(s): AOP-43, SDLP-10 RHR

Proposed references to be provided to applicants during examination: None

Learning Objective: AK2. Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: (CFR: 41.7 / 45.8) AK2.01 Remote shutdown panel: Plant-Specific 4.4* / 4.5*

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 __7__	
	55.43 __5__	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	1	1
Group #	2	1
K/A #	295017 AK3.02	
Importance Rating	3.3	3.2

Proposed Question: 10 / 22

A loss of coolant accident has occurred. The following conditions are present:

Reactor Building Ventilation....................................ISOLATED
Reactor Building Ventilation Exhaust Radiation................1 x 10E5 cpm
"A" Standby Gas Train....................................OPERATING
Reactor Building to Atmosphere Differential Pressure..... negative (-)1.1 inches water
Turbine Building Ventilation ....ISOLATED
Turbine Building Exhaust Radiation..............................3 x 10E4 cpm
Offsite Release ....Above the ALERT Level

Which ventilation system should be reestablished and why?

A. Reestablish the turbine building ventilation to prevent an unmonitored ground level release of radioactivity.
B. Reestablish the turbine building ventilation to filter the turbine building exhaust.
C. Reestablish the reactor building ventilation to prevent an unmonitored ground level release of radioactivity.
D. Reestablish the reactor building ventilation to reduce the reactor building area and equipment temperatures.

Proposed Answer: A. Reestablish the Turbine building ventilation to prevent an unmonitored ground level release of radioactivity.

Explanation (Optional): The turbine building is not a leak tight building. Restarting the turbine building vent will result in an elevated release point for any radioactivity in the building. In addition, EOP-4 states that if reactor building exhaust radiation is greater than 1E4 then isolate reactor building vents.

Technical Reference(s): GE EOP manual page 6-2. There was no basis document for EOP-6. FitzPatrick must very this answer.

Proposed references to be provided to applicants during examination: None

Learning Objective: AK3. Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.5 / 45.6) AK3.02 Plant ventilation 3.3 / 3.5

Question Source: Bank # INPO 6578

Which of the following explains why DEOP 300-2, Radioactive Release Control, directs the operator to restart the Turbine Building Ventilation, if it is shutdown?
to prevent an unmonitored ground level release of radioactivity.
to maintain a positive pressure inside the turbine building.
to reduce the turbine building area and equipment temperatures.

to filter the air in the turbine building before release to the environment.
Reference: ..295017.K3.02

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
	Group #	3	1
	K/A #	295023 AA1.02	
	Importance Rating	2.9	3.1

Proposed Question:11 / 34

The plant is shutdown for a refueling outage with the fuel pool gates installed. Annunciator 09-3-1-9 FUEL POOL COOL & CLN UP TROUBLE alarms. The NLO reports that the spent fuel pool level is slowly dropping and the running fuel pool cooling pump has tripped. Which one of the following methods is available to provide makeup to the spent fuel pool.

A. Align and inject core spray into the reactor cavity.
B. Start the second fuel pool cooling pump to refill the pool.
C. Align condensate transfer to makeup to the skimmer surge tanks.
D. Start a second control rod drive pump to inject into the reactor cavity.

Proposed Answer: C. Align condensate transfer to makeup to the skimmer surge tanks.

Explanation (Optional):
A. The fuel pool gates are installed and because of this, addition of water to the cavity will have no effect on the fuel pool level.
B. Fuel pool level has decreased and has resulted in the alarm. The level in the skimmer surge tanks are the same. The second pump will not start because the level in the skimmer surge tank has fallen below the low low level in the skimmer surge tank.
D. The control rod drive pump will inject into the reactor vessel and will not effect the level in the fuel pool.

Technical Reference(s):SDLP-19, AOP-53

Proposed references to be provided to applicants during examination: None

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: (CFR: 41.7 / 45.6) AA1.02 Fuel pool cooling and cleanup system 2.9 / 3.1

Question Source: Bank # INPO 6671

Which of the following methods/systems is normally used to refill the fuel storage pool?
From the "A" CST via the condensate transfer pumps and refill through the skimmer surge tank(s).

Align Shutdown Cooling and refill via the spent fuel pool diffusers.
Start the second FPCC pump and refill via the spent fuel pool diffusers
Cross connect with Unit 3 FPCC and refill via the spent fuel pool diffusers.

Reference: ..295023.A1.02

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295024 EA2.01	
Importance Rating	_4.2_	_4.4_

Proposed Question: 12 / 8

A major transient has just occurred, no operator actions have been taken. The following plant conditions exist:

Torus pressure ....28 psig
Torus water level ....16 feet
Torus water temperature ....155 °F
DW Pressure ....30 psig
DW Temperature ....290°F
RPV water level ....100 inches
Reactor pressure ....500 psig and dropping

Considering the above conditions, _(1)_ to prevent the _(2)_ from being exceeded.

A. (1) Start Torus venting through SBGT
(2) Pressure Suppression Pressure Limit

B. (1) Start drywell venting through SBGT
(2) Primary Containment Pressure Limit

C. (1) Start Drywell sprays
(2) Drywell Design Temperature

D. (1) Start Torus sprays
(2) SRV Tail Pipe Limit

Proposed Answer: C. (1) Start Drywell sprays
(2) Drywell Design Temperature

Explanation (Optional): A & B These methods should be used during normal operation to maintain the containment pressure below the drywell pressure set point or post accident when conditions in EOP-4 are met. Drywell / Torus pressure conditions are not met to perform post accident containment venting.
C. DW Sprays must be initiated before the drywell design temperature is reached.
D. Initiating torus spray will only move the plant closer to the limit.

Technical Reference(s): EOP-4, Primary Containment Control; EP-6, Post Accident Containment Venting.

Proposed references to be provided to applicants during examination: EOP-4

Learning Objective: EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.10 / 43.5 / 45.13) EA2.01 Drywell pressure 4.2*/ 4.4*

MIT-301.11E, EOP-4

LO 1.03, Identify situations where it is appropriate to enter other procedure concurrently - Task 344169, Spray DW

LO 1.05 Explain basis for any step in the EOP - Task 344169, Spray DW

LO 1.07 Explain the basis and demonstrate the use of all figures associated with EOP-4, Task 344132, Monitor and control DW temperature.

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41 __10__
55.43 __5___

Comments:

The learning objectives do not distinguish between SRO and RO for EOPs. The Objectives state "The operator should be able to ... "

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	__1__
	Group #	_____	__1__
	K/A #	295025 2.2.22	
	Importance Rating	_____	_4.1_

Proposed Question: S 13

Which one of the following conditions would result in a safety limit violation in accordance with Technical Specifications?

A. A high water level in the scram discharge volume which results in a scram. Reactor water level drops to 125.6 inches and water level is being controlled by HPCI.

B. During reactor startup at 30% power, a problem in the EHC Pressure Set causes the bypass valves to open. This results in the reactor pressure dropping to 850 psig.

C. A spurious turbine trip occurs from 100% power. The reactor scram occurred from APRM high flux. Reactor pressure peaked at 1070 psig and is being maintained at 970 psig by the bypass valves.

D. A recirculation pump trips. The reactor operator checks the process computer and finds the minimum fraction of limiting critical power ratio (MFLCPR) to be 0.998.

Proposed Answer: C. A spurious turbine trip occurs from 100% power. The reactor scram occurred from APRM high flux. Reactor pressure peaked at 1070 psig and is being maintained at 970 psig by the bypass valves.

Explanation (Optional):

A. Reactor scram was on a valid signal

B. Reactor did not operate above 25% power at less than 785 psig.

C. TS Bases 1.1.C states that a safety limit violation will be assumed if the scram occurs from other than the expected reactor scram signal. In this case the expected scram signal is 10% closure of the Turbine stop valves.

D. MFLCPR has not exceeded the operating limit MCPR, therefore the safety limit has not been exceeded.

Technical Reference(s): TS 1.1, safety limits

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.2.22 Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2 / 45.2) RO 3.4/SRO 4.1

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:	55.41	_____
	55.43	__2___

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295026 EA2.01	
Importance Rating	_4.1_	_4.2_

Proposed Question: 14 / 26

An ATWS has occurred. All control rods have been inserted using procedure EP-3, "Backup Control Rod Insertion." The following plant conditions exist:

Torus pressure ....5 psig
Torus water level ....14 feet
Torus water temperature ....180 °F
DW Pressure ....5 psig
DW Temperature ....145 °F
RPV water level ....100 inches
Reactor pressure ....1000 psig

Considering the above conditions, you must _(1)_ because the _(2)_ has been exceeded.

A. (1) Perform an Emergency Depressurization
(2) SRV Tail Pipe Limit

B. (1) Perform an Emergency Depressurization
(2) Heat Capacity Temperature Limit

C. (1) Perform RPV Flooding
(2) RPV Saturation Temperature Curve

D. (1) Start Standby Liquid Control
(2) Boron Injection Initiation Temperature

Proposed Answer: B. (1) Perform an Emergency Depressurization
(2) Heat Capacity Temperature Limit

Explanation (Optional):
A The SRV tail pipe limit has not been exceeded.
C. The RPV saturation temperature has not been exceeded.
D. All rods have been inserted reactor power is 0. There is no need to start standby liquid control.

Technical Reference(s): EOP-4, Primary Containment Control

Proposed references to be provided to applicants during examination: EOP-4, EOP-3, Failure to Scram

Learning Objective: EA2. Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) EA2.01 Suppression pool water temperature 4.1* / 4.2*

MIT-301.11E, EOP-4

LO 1.03,	Identify situations where it is appropriate to enter other procedure concurrently
LO 1.05	Explain basis for any step in the EOP
LO 1.07	Explain the basis and demonstrate the use of all figures associated with EOP-4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41 __10__
55.43 __5___

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1___
Group #	__1__	__1___
K/A #	295031 EA1.06	
Importance Rating	_4.4_	__4.4__

Proposed Question: 15 / 10

A small break LOCA has occurred and no operator action has been taken. HPCI is out of service, RCIC is injecting into the reactor vessel and all low pressure ECCS pumps have started from high drywell pressure except the "B" core spray pump which failed to automatically start. One hundred (100) seconds ago the reactor water level was 177 inches, current reactor water level is 75 inches and dropping. Which statement correctly describes the operation of the Automatic Depressurerization System (ADS)?

A. All 7 ADS valves will open in 34 seconds.
B. All 7 ADS valves will open 134 seconds after reactor water level drops to 59.5 inches.
C. All 7 ADS valves will open 134 seconds after the "B" core spray pump is manually started.
D. All 7 ADS valves will open immediately after the "B" core spray pump is manually started.

Proposed Answer: B. All 7 ADS valves will open in 134 seconds after reactor water level decreases to 59.5 inches.

Explanation (Optional):

The following condition are required for ADS to automatically open all 7 valves. Any low pressure ECCS pump running, reactor low 177 <u>and</u> low low low level 59.5 after 134 second timer times out.

Technical Reference(s): SDLP-02J, ADS

Proposed references to be provided to applicants during examination: NONE

Learning Objective: EA1. Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.7 / 45.6) EA1.06 Automatic depressurization system 4.4* / 4.4*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7___
55.43 _____

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295030 EK3.03	
Importance Rating	_3.6_	_3.7_

Proposed Question: 16 / 29

An event has occurred which resulted in torus water level rapidly dropping to 5 feet. Torus level is being maintained at 5 feet. A manual reactor scram has been initiated and an emergency depressurization is in progress using group 2 pressure control systems. Under these plant conditions with torus level at 5 feet, which of the following statements is a valid RCIC operational concern.

A. The RCIC pump suction logic that transfers the RCIC suction to the CST on low Torus level must be defeated.
B. The RCIC system can not be operated in the RPV pressure control mode because it will result in further reduction of torus level.
C. The RCIC suction must remain aligned to the CST to prevent pump suction vortexing.
D. The RCIC system must be tripped to prevent over pressurizing the containment.

Proposed Answer: C. The RCIC suction must remain aligned to the CST to prevent pump suction vortexing. (OP-19 pp. 13)

Explanation (Optional):
A. The RCIC pump suction logic will not transfer on torus low level. This transfer occurs on CST low level.
B. RCIC operation in pressure control mode pumps water from the CST to the CST. OP-19 requires that the pump suction be aligned to the CST prior to starting RCIC in pressure control mode.
D. Containment can not be over pressurized by RCIC turbine exhaust. The containment vent will be able to "keep up" with the pressurization. (MIT-301.11E PP 11)

Technical Reference(s): OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: EK3. Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.5 / 45.6) EK3.03 RCIC operation: Plant-Specific 3.6 / 3.7

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:	55.41	__5__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295031 EK2.04	
Importance Rating	_4.0_	_4.1_

Proposed Question: 17 / 11

A spurious main steam isolation has occurred, all MSIV's have closed. All control rods have inserted and RPV pressure is being controlled 800 to 1000 psig using safety / relief valves. RPV level has just reached 126.5 inches. Which one of the following actions would you expect to occur at this level.

A. A & B Reactor Recirculation Pump Trips.
B. Core Spray System auto start.
C. Reactor Core Isolation Cooling auto start.
D. Standby Gas Treatment System auto start.

Proposed Answer: C. Reactor Core Isolation Cooling auto start.

Explanation (Optional):
A. RR pump will trip at RPV level of 105.4 inches or may already be tripped due to the high pressure from the initial MSIV closure.
B. Core Spray starts at 59.5 inches.
D. SBGT starts at 177 inches.

Technical Reference(s):SDLP-02B, RR and SDLP-02H, RPV Level Instrumentation.

Proposed references to be provided to applicants during examination: None

Learning Objective: EK2. Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: (CFR: 41.7 / 45.8) EK2.04 Reactor core isolation cooling: Plant-Specific. 4.0 / 4.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __7___
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__1__	__1__
	K/A #	295037 EK1.03	
	Importance Rating	_4.2_	_4.4_

Proposed Question: 18 / 12

The reactor failed to scram and the following conditions exist:

RPV water level..........................195 inches
Reactor Power............................18%
Torus Level................................16 ft
Torus water temperature............115 °F
RPV Pressure ....1000 psig
Safety / Relief Valves.................Cycling to control Reactor pressure.
Reactor Recirculation pumps....Tripped
All emergency and normal sources of reactor makeup water are available at full capacity.

In accordance with EOP-3 standby liquid control (SLC) will be initiated based on exceeding the __(1)__. SLC tank level must drop by __(2)__ to ensure that the reactor will stay shut down under all conditions.

A. (1) Heat Capacity Temperature Limit
(2) 22% SLC tank level

B. (1) Heat Capacity Temperature Limit
(2) 46% SLC tank level

C. (1) Boron Injection Initiation Temperature
(2) 22% SLC tank level

D. (1) Boron Injection Initiation Temperature
(2) 46% SLC tank level

Proposed Answer: D. (1) Boron Injection Initiation Temperature
(2) 46% SLC tank level

Explanation (Optional): The Boron Injection Initiation Temperature has been exceeded. At 18% power this correlates to a torus water temperature of 110 °F. 22% SLC tank level will not ensure that the reactor will stay shutdown under all conditions. 46% ensures that a hot / 100% Xenon reactor will be shutdown. 46% tank level will account for the effects of cool down and decay of Xenon.

The heat capacity temperature limit has not been exceeded. At a torus level of 16 feet, RPV pressure of 1000 psig the HCTL is 180 °F Torus water temperature.

Technical Reference(s): EOP-3, "Failure to scram," MIT 301.11d, "Failure to Scram"

Proposed references to be provided to applicants during examination: EOP-3, "Failure to scram

Learning Objective: EK1. Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.8 to 41.10) EK1.03 Boron effects on reactor power (SBLC) 4.2 / 4.4*

MIT 301.11d, EOP-3, "Failure to scram

1.06, Explain the basis of and demonstrate the use of all figures associated with EOP-3

Task 344058, Monitor and control reactor power

Task 344228, Monitor and control torus temperature.

2.04 Explain the reason or purpose for any step in Emergency RPV Depressurization

Question Source: Bank # ________

Modified Bank # ________ (Note changes or attach parent)

New ___X____

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _8-10_

55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295038 EK2.05	
Importance Rating	_3.7_	_4.7_

Proposed Question: 19 / 31

Which one of the following statements correctly describes the color code of the "OFFSITE RAD" box on the EPIC SPDS bar Display.

A. GRAY There are 3 or more invalid inputs in to the OFFSITE RAD" box.
B. GREEN There is an off site release at the ALERT level or lower emergency action level (EAL).
C. MAGENTA There is an off site release at the UNUSUAL EVENT level or higher EAL.
D. RED There is an off site release at the ALERT level or higher EAL.

Proposed Answer: D. RED There is an off site release at the ALERT level or higher EAL.

Explanation (Optional): This box is color coded red or green. Red if Alert or higher and Green is UE or lower.

Technical Reference(s): SDLP-66A, pp 61, EPlan

Proposed references to be provided to applicants during examination: None

Learning Objective: EK2. Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: (CFR: 41.7 / 45.8) EK2.05 †Site emergency plan 3.7 / 4.7*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__1__	__1__
	K/A #	50000 EK3.07	
	Importance Rating	_3.1_	_3.7_

Proposed Question: 20 / 13

The unit has experienced a large break loss of coolant accident which depressurized the reactor. The following conditions exist.

Low Pressure ECCS...............................All pumps are running
Drywell Hydrogen concentration 7%
Drywell Oxygen concentration 7%
Torus Hydrogen concentration 5%
Torus Oxygen concentration...................3%
Offsite release rate ....Will exceed the general emergency release rate

The control room supervisor orders that the drywell be vented and purged per EP-6, through the torus. Based on the Hydrogen & Oxygen concentrations is the action correct and why or why not?

A. No, adequate core cooling is assured, no venting is required until torus Hydrogen is greater than or equal to 6%.
B. No, the drywell can not be vented if the release rate exceeds the general emergency releases rate.
C. Yes, the drywell must be vented to prevent a deflagration and venting through the torus will minimize the radioactive release.
D. Yes, venting the drywell will allow cooler nitrogen to purge the drywell thus slowing down the Zirconium-water reaction which will reduce the hydrogen in the drywell.

Proposed Answer: C. Yes, the drywell must be vented to prevent a deflagration and venting through the torus will minimize the radioactive release.

Explanation (Optional): A & B Based on the Hydrogen and Oxygen concentration the drywell must be vent to prevent a deflagration.
D. Purging the drywell will not have a significant effect on the Zirconium-water reaction.

Technical Reference(s): EOP-4, 4a, EP-6, and MIT-301.11e

Proposed references to be provided to applicants during examination: EOP-4, 4a

Learning Objective: EK3. Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: (CFR: 41.5 / 45.6) EK3.07 Operation of drywell vent 3.1/ 3.7

MIT-301.11E, EOP-4, Primary Containment Control
1.03, Identify situations where it is appropriate to enter other procedures concurrently.
1.05, Explain the basis or purpose for any step in EOP-4.

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__5__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__1__
K/A #	295013 AA2.02	
Importance Rating	_3.2_	_3.5_

Proposed Question: 21 / 20

The NCO2 has been directed to maintain RPV pressure between 800 and 1000 psig using safety relief valves (SRVs). You observe the NCO2 continuously opening and closing the "A" SRV to maintain pressure. Is this an acceptable method of cycling the SRV and what is the basis for this operation?

A. NO. The operator should cycle through the SRVs in the following order A, J, K, G, E, D, C, F, H, L, B to prevent high local pool temperatures that could result in inefficient pool cooling.

B. NO. The operator should cycle through the SRVs in the following order A, J, K, G, E, D, C, F, H, L, B to equally deplete all Nitrogen accumulators.

C. NO. The operator should cycle through the SRVs in the following order A, B, C, D, E, F, G, H, J, K, L to prevent high cyclic fatigue loads, due to chugging, on an individual valve.

D. YES. Cycling only "A" SRV will limit the number of SRV actuations on the other SRVs and minimize the cost required to replace the valve(s) during the next refueling outage.

Proposed Answer: A. NO. The operator should cycle through the SRVs in the following order A, J, K, G, E, D, C, F, H, L, B to prevent high local pool temperatures that could result in inefficient pool cooling.

Explanation (Optional):
B. There is no such requirement.
C. Chugging should not occur if the EOPs are followed.
D. Cycling only one SRV could result in high local pool temperatures that could result in inefficient pool cooling.

Technical Reference(s): MIT 301.11C

Proposed references to be provided to applicants during examination: EOP-2, "RPV Control"

Learning Objective: AA2. Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) AA2.02 Localized heating/stratification 3.2 / 3.5

MIT 301.11C LO EO-1.06

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__10_
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295014 AA1.01	
Importance Rating	_4.0_	_4.1_

Proposed Question: 22 / 6

The plant was operating on a 102% rod line at 85% power and 60% drive flow. Several minutes ago a Hi-Hi level annunciator for the 6B feedwater heater alarmed. All average power range monitors (APRM) are currently at 103% power and all APRM rod block alarms are alarming. The only operator actions taken were to silence and acknowledge annunciators. Which statement correctly assesses the plant condition and gives the correct operator action?

A. A loss of feedwater heating has occurred and power must be rapidly reduced to 85% with reactor recirculation flow.
B. A loss of feedwater heating has occurred and control rods must be inserted to reduce reactor power to 85% power.
C. The APRMs did not initiate an automatic reactor scram, a manual reactor scram is required.
D. The APRMs indicate that core instabilities are present, a manual reactor scram is required.

Proposed Answer: C. The APRMs failed to generator an automatic reactor scram, a manual reactor scram is required. (EOP-2, entry condition)

Explanation (Optional):

A. AOP-62 requires power to be rapidly reduced to 20% below the initial power level. Reduce power to 65%.
B. AOP-62 requires rod insertion until below the 100% rod line. Inserting control rods back to 85% power will only reduce the rod line back to 101%.
C. EOP-2 requires that if reactor has not scrammed then a manual scram is required. The APRM scram set point is 0.58*Wd + 66. This is (.58(62))+66= 101
D. The APRMs are not exhibiting indication of instability. ARPMs are stable not experiencing 10% peak-to-peak oscillations.

Technical Reference(s):AOP-62, EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.7 / 45.6) AA1.01 RPS 4.0/4.1

SDLP-07C, Power Range Monitors
1.07, TS setpoints for (c) APRMs

Question Source:

Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New ___X____

Question History: Last NRC Exam ________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 ____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1_
Group #	__2__	__1_
K/A #	295016 AK3.03	
Importance Rating	_3.5_	_3.7_

Proposed Question: 23 / 21

The shift manager has just ordered a control room evacuation in accordance with AOP-43, "Plant Shutdown From Outside the Control Room." When you arrive at the auxiliary shutdown panel 25 RSP you place the isolation switch for LPCI INBOARD INJECTION VALVE (10MOV-25B) in LOCAL. Reactor water level is 50 inches and reactor pressure is 700 psig. Reactor water level and reactor pressure are dropping. With the isolation switch in LOCAL which statement below describes the operation of the LPCI INBOARD INJECTION VALVE (10MOV-25B) valve as reactor pressure and reactor water level drop.

A. Will automatically open when reactor pressure is less than 450 psig.
B. Will automatically open when the "B" residual heat removal (RHR) pump is started.
C. Must be manually opened from the auxiliary shutdown panel.
D. Must be manually opened at the valve.

Proposed Answer: C. Must be manually opened from the auxiliary shutdown panel.

Explanation (Optional): When the isolation switch for the 10MOV-25B is placed in LOCAL this disables all interlocks and will only allow operation of this valve from the auxiliary shutdown panel. In addition, the valve will not open when the pump is started, the interlock is bases on an ECCS signal and pressure.

Technical Reference(s): AOP-43, SDLP-10 RHR

Proposed references to be provided to applicants during examination: None

Learning Objective: AK2. Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: (CFR: 41.5 / 45.6) AK3.03 Disabling control room controls: 3.5 / 3.7

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 ______

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	_____	__1__
Group #	_____	__1__
K/A #	295023 AK2.07	
Importance Rating	_____	_3.9_

Proposed Question: S 24

The unit was shutdown six days ago to begin a refueling outage. The refueling cavity is flooded and irradiated fuel moves are in progress. Both doors in the drywell personnel air lock have been open to support drywell inspections. Both standby gas treatment trains have just been declared inoperable because they failed to start during a routine surveillance. What actions must be taken in reference to the irradiated fuel moves?

A. Fuel moves must be stopped until both trains of the standby gas treatment system are returned to service because both standby gas trains are needed to maintain reactor building at a negative 0.25 inch water pressure.

B. Fuel moves must be stopped until at least one train of the standby gas treatment system is operable because one standby gas train is capable of maintaining the reactor building at a negative 0.25 inch water pressure.

C. Fuel moves must be stopped until one drywell personnel air lock door is secure closed to reestablish primary containment.

D. Fuel moves may continue for the next seven days. If at least one train of standby gas treatment is not returned to an operable condition within this seven day period then fuel movement must be stopped.

Proposed Answer: B. Fuel moves must be stopped until one train of the standby gas treatment system is operable because one standby gas train is capable of maintaining the reactor building at a negative 0.25 inch water pressure.

Explanation (Optional):

A. TS 3.7.B Basis only one train of SGT is required to maintain a 0.25 inch water pressure.

C. Primary containment is not required in this condition.

D. TS 3.7.B.2 requires that no irradiated fuel movement take place unless at least one train of SGT available.

Technical Reference(s): TS

Proposed references to be provided to applicants during examination: None

Learning Objective: AK2. Knowledge of the interrelations between REFUELING ACCIDENTS and the following: (CFR: 41.7 / 45.8) AK2.07 Standby gas treatment/FRVS 3.6 / 3.9

Question Source:

Bank # ________

Modified Bank # ________ (Note changes or attach parent)

New ___X____

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7___
55.43 __2___

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__1__	__1__
K/A #	295025 K1.04	
Importance Rating	_3.6_	_3.9_

Proposed Question: 25 / 9

The plant has just scrammed after an extended full power run. The feedwater pumps are maintaining RPV level and RPV pressure is 970 psig using the bypass valves. NCO2 has just reported a trip of the "A"&"B" circulating water pumps and has taken appropriate action for the loss of these pumps in accordance with AOP-1, "Scram." If no additional operator action is taken how will RPV pressure respond?

A. will drop from 970 psig as decay heat drops

B. will remain at or below approximately 970 psig

C. will rise to 1135 psig

D. will rise to 1145 psig

Proposed Answer: C. will rise to 1135 psig

Explanation (Optional): The "C" Circ water pump was lost on the scram. When the A&B circ water pump is lost AOP-1 directs closing the MSIVs. If the MSIVs are closed then RPV pressure will be controlled at the lowest relief valve setting 1135psig. Since one relief valve will pass about 8% total steam flow only one will be open and the pressure will remain at that setting until it decays away at a later time.

Technical Reference(s):SDLP 02J pp 23 and SDLP-94C figure 8

Proposed references to be provided to applicants during examination: None

Learning Objective: EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: (CFR: 41.8 to 41.10) EK1.04 Decay heat generation 3.6 / 3.9

Question Source: Bank # FitzPatrick Requalification 753

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __8__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__1__
Group #	____	__1__
K/A #	295014 AK2.08	
Importance Rating	____	_3.5_

Proposed Question: S26

The plant is operating at 55% power and 50% core flow when a peripheral control rod (02-27) is observed drifting out of the core. In accordance with AOP-27, "Control Rod Drift," the operator is FIRST required to insert the control rod by __(1)__ . While being inserted the control rod stops at position 24. Further investigation determines that the control rod is stuck at position 24 and the cause of the failure is unknown. The plant __(2)__ , in accordance with technical specifications.

A. (1) performing an individual rod scram
(2) may continue to operate provided a new shutdown margin calculation is performed

B. (1) using the rod emergency in notch override switch
(2) must be brought to cold shutdown in 24 hours

C. (1) inserting a manual reactor scram
(2) must be shutdown and can not be restarted unless investigation has shown that the cause of the failure was not a failed control rod drive collet housing

D. (1) using the rod emergency in notch override switch
(2) can not be restarted until the NRC approves restart.

Proposed Answer: B. (1) using the rod emergency in notch override switch
(2) must be brought to cold shutdown in 24 hours

Explanation (Optional):
A. The first method of inserting the control rod is by using the control rod emergency in switch, not individual scram switch. In addition, if a control is "STUCK" then the reactor must be shutdown in 24 hours. There is no provision to allow continued operation.
B. Correct
C. A manual reactor scram is required if more than one control rod is drifting. There is only one control rod drifting.
D. The NRC would not be required to authorize startup in this case.

In addition, this is a peripheral rod and little power change is expected.

Technical Reference(s): AOP-27, control rod drift; AOP-32, "UNEXPLAINED/UNANTICIPATED REACTIVITY CHANGE*" TS 3.3.A.2

Proposed references to be provided to applicants during examination: None

Learning Objective: AK2. Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: (CFR: 41.7 / 45.8) AK2.08 RMCS: 3.4/3.5

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	__7__
	55.43	_2, 5_

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	_____	__1__
Group #	_____	__2__
K/A #	2950001 AK1.01	
Importance Rating	_3.5_	_3.6_

Proposed Question: 27 / 14

The plant is operating at 100 % when I&C inadvertently trips both reactor recirculation pumps while working on the ARI/RPT logic. Where will the reactor be on the Power-Flow Map and what actions are required.

A. The reactor will be in the Power-flow Map BUFFER ZONE. Monitor nuclear instrumentation, for indications of thermal-hydraulic instability.
B. The reactor will be in the Power-flow Map BUFFER ZONE. Manually scram the reactor.
C. The reactor will be in the Power-flow Map EXCLUSION ZONE. Manually insert control rods to exit the exclusion zone.
D. The reactor will be in the Power-flow Map EXCLUSION ZONE. Manually scram the reactor.

Proposed Answer: D. The reactor will be in the Power-flow Map EXCLUSION ZONE. Manually scram the reactor

Explanation (Optional): D. The trip of both RWR pumps will result in the plant being at about 50% power on the natural circulation line of the power to flow map. AOP-8 requires a manual scram when both RWR pumps are tripped.

A. The reactor not be in the buffer zone.
B. The reactor will not be in the buffer zone.
C. The actions listed in C are correct if both RR pumps have not tripped. However, the first step in AOP-8 requires a manual scram if both RWR pumps trip.

Technical Reference(s): TS Basis 3.5.J, AOP-8

Proposed references to be provided to applicants during examination: None

Learning Objective: AK1. Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.8 to 41.10) AK1.01 Natural circulation 3.5 / 3.6

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:	55.41	__8__
	55.43	__5__

Comments:

Examination Outline Cross-reference: Level		RO	SRO
	Tier #	__1__	__1__
	Group #	__2__	__2__
	K/A #	295002 AK2.07	
	Importance Rating	_3.1_	_3.1_

Proposed Question: 28 / 15

The plant is operating at 100 % power when annunciator 09-6-1-29 CNDSR VAC LO alarms. Condenser vacuum has slowly dropped to 25 inches of Hg and generator output has dropped by 3 MWe. A NLO identified a small tear in the expansion boot between the condenser and LP turbine hood. How has the offgas flow changed (prior to any operator action) and what actions, in addition to a power reduction, will the operator take in response to this event.

A. Offgas flow has dropped, trip hydrogen addition and start the condenser air removal pumps.
B. Offgas flow has dropped, trip the turbine, scram the reactor and close the main steam line isolation valves.
C. Offgas flow has risen, trip hydrogen addition and start the condenser air removal pumps.
D. Offgas flow has risen, place the spare steam jet air ejectors in service.

Proposed Answer: D. Off gas flow has INCREASED, place the spare steam jet air ejector in service.

Explanation (Optional): A. & B. The off gas flow will increase not decrease with a condenser boot tear. There is more non Condensible gas entering the condenser and the vacuum decreases.

C. Condenser air removal pumps discharge to the 1.75 minute holdup pipe, which is not designed for explosion pressure. For this reason, operation of condenser air removal pumps is not permitted if reactor power is greater than 5%. Power is greater than 5%.

Technical Reference(s): AOP-31, loss of condenser vacuum

Proposed references to be provided to applicants during examination: None

Learning Objective: AK2. Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: (CFR: 41.7 / 45.8) AK2.07 Offgas system 3.1 / 3.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X___

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__2__	__2__
	K/A #	295004 Ak3.02	
	Importance Rating	_2.9_	_3.3_

Proposed Question: 29 / 17

The plant is operating at 100% with operators performing AOP-22, "DC Power System A Ground Isolation," due to a ground on the "A" station battery. The next breaker to be opened is the supply for the 10700 bus breaker Control Power. When this breaker is OPENED, which one of the following statements correctly describe the effects or actions that this will have on the 10700 bus?

A. All 10700 bus breaker protection trips will operate normally because the bus logic power has automatically swapped to "B" 125 VDC.

B. All 10700 bus breaker red / green position indicating lights will still indicate breaker positions because the logic power has automatically swapped to "B" 125 VDC.

C. All 10700 bus breakers will open if originally closed due to a loss of "A" 125 VDC control power.

D. All 10700 bus breakers will lose red / green position indicating lights because the breakers have lost "A" 125 VDC control power.

Proposed Answer: D. All 10700 bus breakers will lose red / green position indicating lights because the breakers have lost "A" 125 VDC control power.

Explanation (Optional): A. & B. There is no automatic swap of 125 VDC control power on bus 10700, and the red and green lights will be lost when 125 VDC is lost.

C. The breakers will not automatically open on loss of DC control power.

Technical Reference(s): SDLP-71B, AOP-22, OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: AK3. Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.5 / 45.6) AK3.02 Ground isolation/fault determination 2.9 / 3.3

Question Source: Bank # FitzPatrick Requalification bank 0869

The plant is operating at 100% with operators performing AOP-22, DC POWER SYSTEM A GROUND ISOLATION due to a ground on the "A" station battery. The next breaker to be opened is the supply for 10700 BKR Control Power (71DCA3 Crkt 24). When this breaker is OPENED, Which one of the following statements will occur?

a) All 10700 bus breaker protection trips will operate normally.

b) All 10700 bus breakers will open if originally closed.

c) All 10700 bus breakers can be tripped locally if closed.

d) All 10700 bus breaker position indication lights (red and green) will still indicate breaker positions.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__2__	__2__
	K/A #	295008 AA1.01	
	Importance Rating	_3.7_	_3.7_

Proposed Question: 30 / 18

The reactor protection system initiated a scram, all blue scram lights are lit on the full core display. However, little rod motion occurred, power reduced to 95% and the plant remains stable with the following indications:

Reactor Power............................95%
"A" RWR Pump Speed..............90%
"B" RWR Pump Speed..............91%
Mode Switch..............................SHUTDOWN
Alternate Rod Insertion...............INITIATED
Turbine......................................ON LINE
Reactor Pressure........................1035 psig
Reactor Level.............................200 inches
Drywell Temperature..................130 °F
Drywell Pressure........................2.2 psig
Torus Water Temperature..........75 °F
Torus Pressure ....0.3 psid

What operator actions must be taken and why?

A. Inject standby liquid control because the boron injection temperature has been exceeded.
B. Immediately trip both RWR pumps to achieve a rapid power reduction.
C. Reduce recirculation flow to minimum then trip both RWR pumps to prevent the turbine from tripping.
D. Vent the scram air header to open the scram inlet and outlet valves.

Proposed Answer: C. Reduce recirculation flow to minimum then trip both RWR pumps to prevent the turbine from tripping.

Explanation (Optional):
A. The Boron injection temperature has not been exceeded.
B. In this case the RWR should be run to minimum speed to prevent a RPV high level transient from tripping the turbine and forcing all the heat load into containment.
D. All the scram inlet and outlet valves are open as indicated by the blue lights on the full core display.

Technical Reference(s): EOP-3, MIT 301.11d, page 5

Proposed references to be provided to applicants during examination: EOP-3

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL: (CFR: 41.7 / 45.6) AA1.01 Reactor water level control: Plant-Specific 3.7 / 3.7

MIT 301.11d
1.07 Explain the basis or purpose for any step in EOP-3

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	295018 A2.01	
Importance Rating	_3.3_	_3.4_

Proposed Question: 31/ 23

The "B" & "D" emergency diesel generators (EDGs) are tagged out of service. A loss of coolant accident and a loss of offsite power have just occurred. "A" & "C" EDGs are operating and required for core cooling. The ESW LOW FLOW annunciator is alarming due to a failure of the "A" emergency service water (ESW) pump. The "A" & "C" EDGs have a jacket water temperature of 190°F and rising. How will the "A" & "C" EDG respond to this condition and what operator actions, if any, are required? The "A" & "C" EDGs...

A. will automatically trip when the jacket water temperature reaches 205°F, attempt to start the emergency service water (ESW) pump.
B. must be manually tripped before the jacket water temperature reaches 205°F, have the control room shutdown the "A" & "C" EDGs.
C. will continue to run at jacket water temperatures above 205°F, align the fire protection system to supply the "A" & "C" EDGs.
D. high jacket water temperature trip must be bypassed before reaching 205°F, install jumpers to bypass the high temperature trips on the "A" & "C" EDGs.

Proposed Answer: C. will continue to run at jacket water temperatures above 205°F, align the fire protection system to supply the "A" & "C" EDGs.

Explanation (Optional):
A. If a LOCA signal is present the EDG will not trip on high jacket water temperature.
B. The EDG should not be shutdown if they are required for core cooling. In addition, at higher temperature the EDG will continue to run but at reduced load. The EDGs can not be shutdown from the control room.
C. Correct - ARP 93ECP-A-12 and OP-22
D. The trip is automatically bypassed under LOCA conditions.

Technical Reference(s): ARP 93ECP-A-12, OP-22

Proposed references to be provided to applicants during examination: None

Learning Objective: AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.10 / 43.5 / 45.13) AA2.01 Component temperatures 3.3 / 3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	_10__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level		RO	SRO
	Tier #	_____	__1__
	Group #	_____	__2__
	K/A #	295019 2.4.11	
	Importance Rating	_____	_3.6_

Proposed Question: S32

The plant was operating at 100% power with a normal plant line up. Several minutes ago a transient occurred. The following annunciators are some of the annunciators that are alarming.

SERV AIR HDR PRESS LO
SERV AIR HDR ISOL VLV CLOSED
SCRAM AIR HDR PRESS HI OR LO
ROD DRIFT
RWM ROD BLOCK
CNDSR VAC LO

What system is causing the degrading plant conditions and what operator action is required.

A. The service air system has degraded and the reactor water recirculation (RWR) pumps must be tripped because of a loss of cooling water to the pump motor.
B. The instrument air system has degraded and a manual reactor scram is required because at least one control rod has started to drift.
C. The running control rod drive pump has tripped and a manual reactor scram is required because at least one control rod has started to drift.
D. Condenser vacuum has degraded due to a tear in the turbine exhaust expansion boot. NLOs must be dispatched to look for the leak.

Proposed Answer: B. The instrument air system has degraded and a manual reactor scram is required because at least one control rod has started to drift.

Explanation (Optional):
A. The RWR do not have to be tripped because the cooling water valves to these pumps do not close on a loss of instrument air.
C. The running control rod drive pump has not tripped. The loss of instrument air will close the pump discharge FCV, not trip the pump.
D. The loss of vacuum is caused by the condenser water box air admission valve opening due to the loss of instrument air, not a tear in the expansion boot.

Technical Reference(s): AOP-12 LOSS OF INSTRUMENT, AIR*; AOP-31 LOSS OF CONDENSER VACUUM; OP-39 BREATHING, INSTRUMENT, AND SERVICE AIR SYSTEM

Learning Objective: 2.4.11, knowledge of abnormal response procedures

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	_____
	55.43	__5__

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__2__	__2__
	K/A #	295020 AA2.01	
	Importance Rating	_3.6_	_3.7_

Proposed Question: 33 / 24

The Unit is operating at 100% power when a spurious loss of RPS bus "A" occurs. The following conditions are present after the loss of the "A" RPS bus.

Drywell Temperature ....130°F
Drywell Pressure..2.1 psig
Torus Water Temperature ....72°F
Torus Pressure ....0.8 psig
Torus Level ....13.98 Feet
Drywell and Torus Oxygen Concentration ....2.1 volume percent

Based on these conditions what operator actions are required after the "A" RPS bus is restored per AOP-59, "Loss of RPS Bus A Power."

A. Reopen the reactor building closed loop cooling Drywell Cooler "A" Inlet and outlet valves to ensure that the drywell pressure remains below 2.7 psig.
B. Vent the drywell through standby gas to ensure that the drywell pressure remains below 2.7 psig.
C. Vent the torus to establish drywell to torus differential pressure within the Technical Specification required value.
D. Start drywell makeup using CAD Train A to maintain the oxygen concentration within the Technical Specification required value.

Proposed Answer: C. Vent the Torus to establish drywell to torus differential pressure within the Technical Specification required value.

Explanation (Optional):

A. The Drywell Cooler valves do not go closed on a loss of RPS.
B. By venting the Drywell through SBGT the drywell to torus dP will be reduced further.
C. Differential pressure is 1.3, TS requires > 1.7 psid.
D. The Oxygen concentration is allowable by TS (less than 4.0 volume Percent)

Technical Reference(s): TS 3.7, OP-37, Containment Atmosphere Dilution System, AOP-59 Loss of RPS bus A Power

Proposed references to be provided to applicants during examination: None

Learning Objective: AA2. Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.10 / 43.5 / 45.13)
AA2.01 Drywell/containment pressure 3.6 / 3.7

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__5___
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__3__	__2__
K/A #	295021	
Importance Rating	_3.5_	_3.5_

Proposed Question: 34 / 33

The plant is in shutdown with the "B" residual heat removal system operating in shutdown cooling mode. The following plant conditions exist.

RPV pressure ....0 psig
RPV level ....270 inches
RPV head ....INSTALLED
Coolant temperature ....100°F

"A" & "B" Reactor Water Recirculation (RWR) pumps ....OFF
"B" Reactor Recirculation Pump Suction Valve (02MOV-43B)...................OPEN
"B" Reactor Recirculation Pump Discharge Valve (02MOV-53B) ....CLOSED

Maintenance is troubleshooting the "B" Reactor Recirculation Pump Discharge Valve (02MOV-53B) when the valve inadvertently opens. What effect does this have on shutdown cooling.

A. This will create a large drain path to the torus and will reduce the RPV water level until the group II isolation occurs on RPV low water level.
B. This will result in a loss of shutdown cooling and insufficient natural circulation.
C. This will result in a reduction of shutdown cooling because some of the shutdown cooling flow will bypass the reactor core.
D. This will result in a loss of valid reactor coolant temperature indication because of insufficient natural circulation.

Proposed Answer: C. This will result in a reduction of shutdown cooling because some of the shutdown cooling flow will bypass the reactor core.

Explanation (Optional): Opening the RWR pump discharge valve will allow the RHR shutdown cooling flow to bypass the core. However, the RPV water level is greater than 234.5 which will ensure good natural circulation and temperature indication is available.

Technical Reference(s): SDLP-02H, SDLP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: AA1. Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.7 / 45.6) AA1.02 RHR/shutdown cooling 3.5 / 3.5

Question Source:

Bank # ________
Modified Bank # ________ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__

55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__2__	__2__
	K/A #	295022 Ak3.02	
	Importance Rating	_2.9_	_3.1_

Proposed Question: 35 / 25

Five minutes ago both CRD pumps tripped. This condition will result in a loss of __(1)__ and may result in __(2)__ .

A. (1) Cooling water flow to the CRD
(2) degradation of the Graphitar seals

B. (1) Charging water flow to the CRD
(2) loss of scram capability

C. (1) Drive water flow to the CRD
(2) failure of the chromel / alumel temperature sensor in the position indication probe

D. (1) Exhaust water flow from the CRD
(2) failure of the collet housing from inter-granular stress corrosion cracking

Proposed Answer: A. (1) Cooling water flow to the CRD
(2) degradation of the Graphitar seals

Explanation (Optional): At elevated temperatures the Graphitar seals become brittle and can result in breakdown and increased scram times.

Technical Reference(s): SDLP-03A, AOP-69, Control Rod Drive Trouble

Proposed references to be provided to applicants during examination: None

Learning Objective: AK3. Knowledge of the reasons for the following responses as they apply to LOSS OF CRD PUMPS: (CFR: 41.5 / 45.6) AK3.02 CRDM high temperature 2.9 / 3.1

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5___
55.43 _____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__2__	__2__
	K/A #	295028 EK2.02	
	Importance Rating	_3.2_	_3.3_

Proposed Question: 36 / 27

A LOCA has just occurred and all plant systems functioned as designed. Based on the following list of plant parameters what reactor water level indication may be used by the operator.

Drywell Pressure ....35 psig
Drywell Instrument Run Temperatures ..320° F
Torus Pressure ....33 psig
Torus Water Temperature ....150° F
Reactor Pressure ....100 psig

A. Refuel Zone Level Indicating 200 inches
B. Narrow Range Level Indicating 164.5 inches
C. Fuel Zone Level Indicating negative (-)100 inches
D. RPV water level can not be determined

Proposed Answer: C. Fuel Zone Level Indicating -100 inches

Explanation (Optional):
A. Level is NOT above its minimum usable indication level.
B. Level is NOT above its minimum usable indication level.
C. Correct
D. The Fuel Zone Level indication is on scale and useable based on instrument run temperatures.

Technical Reference(s): EOP-2, "RPV Control"

Proposed references to be provided to applicants during examination: EOP-2, "RPV Control"

Learning Objective: EK2. Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: (CFR: 41.7 / 45.8) EK2.02 Components internal to the drywell 3.2 / 3.3

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __7___
55.43 ______

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__2__	__2__
K/A #	295029 EK1.01	
Importance Rating	_3.4_	_3.7_

Proposed Question: 37 / 28

The control room supervisor has determined that torus water level and RPV pressure can not be maintained below the SRV Tail Pipe Level Limit. Why is an emergency depressurization required?

A. The SRV vacuum breakers will be submerged and will not limit the dynamic forces created by steam condensation in the SRVs tail pipes.
B. The high torus level will flood the HPCI and RCIC turbine exhaust lines and resulting in a common mode failure of emergency high pressure injection.
C. SRV operation could result in failure of the SRV tail pipes and direct containment pressurization.
D. Containment overpressurization could occur from the torus to drywell vacuum breakers being submerged.

Proposed Answer: C. SRV operation could result in failure of the SRV tail pipes and direct containment pressurization.

Explanation (Optional):
A The SRV vacuum breakers are located in the drywell and will not be submerged.
B. There are check valves in the HPCI and RCIC turbine exhaust lines that would prevent flooding.
D. Containment over pressure would not occur. Most of the non condensible is the nitrogen and to displace the nitrogen you would need a LOCA not a blowdown. In addition the containment vent would be able to keep up with the increased pressure during a blowdown.

Technical Reference(s): EOP-4, Primary Containment Control; EP-6, Post Accident Containment Venting.

Proposed references to be provided to applicants during examination: EOP-4

Learning Objective: EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL: (CFR: 41.8 to 41.10) EK1.01 Containment integrity 3.4 / 3.7

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:	55.41	__8__
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	__1__
	Group #	_____	__2__
	K/A #	295032 K2.01	
	Importance Rating	_3.5_	_3.6_

Proposed Question: S38

The plant is operating at 100% power. There are no Technical Specification (TS) Limiting Conditions for Operation (LCO) in effect. The shift manager has just approved work on the reactor core isolation cooling (RCIC) ventilation system. This work removes the RCIC ventilation fans, closes the fan dampers and installs temporary house fans, using non emergency power, to be placed inside the door ways. Must a TS LCO be entered for this work?

A. No. A TS LCO is not required because the RCIC ventilation dampers are in the fail safe position for a fire.

B. No. A TS LCO is not required because the RCIC ventilation system is not required for the operability of RCIC.

C. Yes. A TS LCO is required for RCIC because the normal RCIC ventilation, powered by emergency power, is inoperable.

D. Yes. A TS LCO is required for secondary containment because the RCIC ventilation is inoperable.

Proposed Answer: C. Yes. A TS LCO is required because the normal RCIC ventilation, powered by emergency power, is inoperable.

Explanation (Optional): TS Definitions require that all necessary attended support systems are operable to support the RCIC system. Since the normal, emergency powered RCIC ventilation is not operable, RCIC is not operable. RCIC may be available but not operable. If RCIC is inoperable a 7 day LCO is required.(TS 3.5.E). The RCIC ventilation line up in this question would not effect the secondary containment ventilation.

Technical Reference(s): TS 3.5.E, Definitions

Proposed references to be provided to applicants during examination: None

Learning Objective: EK2. Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: (CFR: 41.7 / 45.8)
EK2.01 Area/room coolers 3.5 / 3.6

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:	55.41	____
	55.43	__2__

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	__1__
	Group #	_____	__2__
	K/A #	295032 Ek2.01	
	Importance Rating	_3.5_	_3.6_

Proposed Question: S39

Several minutes ago the reactor water cleanup (RWCU) suction pipe failed between the supply inboard isolation valve (MOV-15) and supply outboard isolation valve (MOV-18). The inboard isolation valve failed to close and all other plant equipment operated as designed. The following plant conditions are present.

Reactor power...............0% ....(manual scram)
Reactor level ....200 inches
Reactor pressure ....700 psig ....(Emergency Depressurization in progress)

RWCU

Heat Exchanger Room Temperature ......200°F
"A" RWCU Pump Room Temperature ....220°F
"B" RWCU Pump Room Temperature ....200°F

Heat Exchanger Area Radiation ....1020 mr/hr
"A" RWCU Pump area ....1150 mr/hr

The shift manager declared a site area emergency and all notifications were made. The site area emergency declaration was made because plant conditions represent

A. a loss of the containment barrier and a loss of the reactor coolant system barrier.
B. an actual or imminent substantial core degradation or melting with a potential for loss of containment integrity.
C. a loss of the primary containment with a loss of the secondary containment.
D. a loss of the containment barrier and a loss of the reactor coolant system barrier with imminent substantial core degradation or melting.

Proposed Answer: A. a loss of the containment barrier and a potential loss of the reactor coolant system barrier.

Explanation (Optional):
A. For the purpose of the EALs the secondary containment is not considered a fission product barrier (IAP2.2, page 7). Therefore, primary containment is failed.
B. This condition does not represent substantial core degradation because all other plant equipment functioned as designed.
C. For the purpose of the EALs the secondary containment is not considered a fission product barrier (IAP2.2, page 7)
D. This condition is not true because adequate core cooling will be able to be maintained through core submergence.

Technical Reference(s): Emergency Plan implementing Procedures / Volume 2, IAP-2.

Proposed references to be provided to applicants during examination: EOP-5/6

Learning Objective: EK3. Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: (CFR: 41.5 / 45.6) EK3.05 Emergency plan 3.6 / 4.5*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 __4__

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__2__	__2__
	K/A #	295034 EA1.01	
	Importance Rating	_3.8_	_3.8_

Proposed Question: 40 / 30

The "A" reactor water cleanup (RWCU) pump has a seal leak. All reactor building temperature and radiation levels are normal except for the following. The RWCU pump area radiation monitor is reading 75 mr/hr and rising. The reactor building vent exhaust radiation monitors 17RM-452A & B are reading 3 x 10E3 and rising slowly. In addition to monitoring the RWCU pump area temperature and radiation levels what other actions must be taken.

A. Isolate the "A" RWCU pump and if reactor building vent radiation exceeds 1 x 10E4 then ensure SBGT starts and the reactor building has isolated.
B. Isolate the "A" RWCU pump, immediately isolate the reactor building and start SBGT.
C. Immediately isolate the reactor building and start the standby gas treatment system.
D. Enter EOP-2, "RPV Control."

Proposed Answer: A. Isolate the "A" RWCU pump and if reactor building vent radiation exceeds 1 x 10E4 then ensure SBGT starts and the reactor building has isolated.

Explanation (Optional):
B. EOP-5 does not require the vents to be isolated until 1X10E4 is reached and immediately isolating the RB vents will degraded the plant further because cooling to plant equipment will be further decreased.
C. EOP-5 does not require the vents to be isolated until 1X10E4 is reached and immediately isolating the RB vents will degraded the plant further because cooling to plant equipment will be further decreased.
D. Current conditions do not require this procedure to be entered.

Technical Reference(s): EOP-5, SDLP 12, 17, 01B

Proposed references to be provided to applicants during examination: EOP-5

Learning Objective: EA1. Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: (CFR: 41.7 / 45.6) EA1.01 Area radiation monitoring system 3.8 / 3.8

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__3__	__2__
	K/A #	295035 A2.01	
	Importance Rating	_3.8_	_3.9_

Proposed Question: 41 / 35

During alignment of reactor building ventilation the operator notices that reactor building to atmosphere differential pressure is positive and rising. At what differential pressure will the Reactor Building ventilation isolate and what operator actions will be required to maintain secondary containment integrity?

A. At + 1 inch of water the reactor building isolation occurs, verify the standby gas treatment system has automatically started.
B. At + 1 inch of water the reactor building isolation occurs, manually start the standby gas treatment system.
C. At + 4 inches of water the reactor building isolation occurs, verify the standby gas treatment system has started.
D. At + 4 inches of water the reactor building isolation occurs, manually start the standby gas treatment system.

Proposed answer: B. At + 1 inch of water the reactor building isolation occurs, manually start the standby gas treatment system.

Explanation (Optional): At +1 inch water the reactor building isolates. Standby gas does not automatically start on this isolation so it must be manually started.

Technical Reference(s): OP-51A REACTOR BUILDING VENTILATION AND COOLING SYSTEM* and SDLP-01B & 66

Proposed references to be provided to applicants during examination: None

Learning Objective: EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.8 to 41.10) EA2.01 Secondary containment pressure: 3.8 / 3.9

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __9__
55.43 _____

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__1__	__1__
Group #	__3__	__2__
K/A #	295036 2.1.7	
Importance Rating	_3.7_	_4.4_

Proposed Question: 42 / 36

While the plant is operating at 100% power, a fault occurs in the fire protection system resulting in discharge into the east and west crescent areas. Water levels in both crescent areas are 20 inches and rising. Choose the statement below that describes the operator action that is required.

A. Scram the reactor.
B. Commence a reactor shut down.
C. Isolate sump discharge to radwaste storage tanks.
D. Perform an EMERGENCY DEPRESSURIZATION in accordance with EOP-2.

Proposed Answer: B. Commence a reactor shut down.

Explanation (Optional): The water levels in EOP-5, "Secondary Containment Control" have been exceeded and there is NOT a primary system discharging into the area, therefore a normal shutdown is directed.

Technical Reference(s): EOP-5, MIT 301.11F

Proposed references to be provided to applicants during examination: EOP-5

Learning Objective: 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation - RO 3.7/SRO 4.4
MIT-301.11F, EO 5.07

Question Source: Bank # FitzPatrick Requalification Bank 20005219B01C Rev.2
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __10_
55.43 _____
LO MIT-301.11F, EO 5.07

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__1__	__1__
	Group #	__2__	__2__
	K/A #	600000 A2.16	
	Importance Rating	_3.0_	_3.5_

Proposed Question: 43 / 32

The plant is operating at a 100% power, with a normal plant line up when a fire in the Control Room causes an immediate evacuation, such that none of the required immediate actions can be taken. All operator actions must be taken from outside the Control Room. What operator actions must be taken to scram the reactor and what other actions will result?

A. Both the RPS MG set output breaker AND RPS alternate feeder breaker must be OPENED on the A & B RPS system. This will result in a scram only.

B. The A & B RPS MG set output breakers must be OPENED. This will result in a scram only.

C. The A & B RPS MG set output breakers must be OPENED. This will result in a scram and group 1 isolation only.

D. The A & B RPS MG set output breakers must be OPENED. This will result in a scram, Group I isolation, and Group II isolation.

Proposed Answer: D. The A & B RPS MG set output breakers must be OPENED. This will result in a scram, Group I isolation, and Group II isolation.

Explanation (Optional):

A. Only the RPS MG Set breakers need be open to initiate a scram. Both breakers do not need to be opened.
B. This will result in more than a scram
C. If the RPS system losses power PCIS will actuate as well.
D. Correct.

Technical Reference(s): SDLP-16C, AOP-43 PLANT SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

Proposed references to be provided to applicants during examination: None

Learning Objective: AA2 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: AA2.12 Location of vital equipment within fire zone 3.1 / 3.5 A2.16 Vital equipment and control systems to be maintained and operated during a fire.

Question Source: Bank # FitzPatrick Requalification Question 126

A fire in the Control Room causes immediate evacuation, such that none of the required immediate actions can be taken. All actions must be taken from outside the Control Room. Which of the following statements is correct?

A. In order to scram the reactor the RPS A and B MG set output breakers AND RPS alternate feeder breakers must be opened.

B. De-energizing RPS A and B by opening both MG set output breakers will cause a scram ONLY.

C. De-energizing RPS A and B by opening both MG set output breakers will cause a scram, Group I isolation, and Group II isolation.

D. De-energizing RPS A and B by opening both MG set output breakers will cause a scram and Group I isolation ONLY.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __8__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__2__
Group #	_____	__1__
K/A #	261000 A3.01	
Importance Rating	_3.2_	_3.3_

Proposed Question: S44

The plant is operating at 100% power with the "A" standby gas treatment train out of service when the "B" RPS bus is lost. The following conditions are present five minutes after the loss of the "B" RPS bus.

Drywell Temperature ....130°F
Drywell Pressure... ....2.0 psig
Torus Water Temperature ....72°F
Torus Pressure ....0.1 psig
Torus Level .. ..13.98 Feet
Reactor Building to Outside dP.. 0.1 inches water
SGTS Train B ....ON with a flow rate of 1000 cfm
Drywell and Torus Oxygen Concentration ....2.1 volume percent

Based on these conditions what operator actions are required?

A. Reopen the reactor building closed loop cooling Drywell Cooler B Inlet valve to ensure that the drywell pressure remains below 2.7 pisg.
B. Declare the "B" SBGT inoperable and enter the Technical Specification Limiting Condition for Operation.
C. Vent the Torus to maintain drywell to torus differential pressure within the Technical Specification allowable values.
D. Start drywell makeup using CAD Train A to maintain the oxygen concentration within the Technical Specification allowable value.

Proposed Answer: B Declare the "B" SBGT inoperable and enter the TS Limiting Condition for Operation

Explanation (Optional):

A. The Drywell Cooler valves do not go closed on a loss of RPS.
B. SBGT is malfunctioning and because of the low flow it can not maintain 0.25 inches water dP in secondary containment.
C. Differential pressure is 2.0, allowable by TS (> 1.7 psid)
D. The Oxygen concentration is allowable by TS (less than 4.0 volume Percent)

Technical Reference(s): TS 3.7, OP-37, Containment Atmosphere Dilution System, AOP-60 Loss of RPS bus B Power

Proposed references to be provided to applicants during examination: None

Learning Objective: A3. Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: (CFR: 41.7 / 45.7) A3.01 System flow 3.2 / 3.3

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 ____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	_2_
	Group #	_____	_1_
	K/A #	202002 A2.09	
	Importance Rating	_3.1__	_3.3_

Proposed Question:S45

The plant is at 80% power. Due to a small speed oscillation, the "A" reactor recirculation pump scoop tube was "locked up," per OP-27, "Recirculation System." With the "A" reactor recirculation pump scoop tube "locked up," the "B" feedwater pump trips, which results in reactor water level dropping to 195 inches. Based on these conditions __(1)__ and procedure __(2)__ will be used.

A. (1) only the "B" reactor recirculation pump will run back to 44%
(2) AOP-1, "Reactor Scram."

B. (1) both the A & B reactor recirculation pumps will run back to 44%
(2) AOP-42, "Feedwater Malfunction (Lowering Feedwater Flow)"

C. (1) only the "A" reactor recirculation pump will trip
(2) AOP-8, "Loss or Reduction of Reactor Coolant Flow"

D. (1) both the A & B reactor recirculation pumps will trip
(2) AOP-8, "Loss or Reduction of Reactor Coolant Flow"

Proposed Answer: B. (1) Both the A & B reactor recirculation pumps will run back to 44%
(2) AOP-42, "Feedwater Malfunction (Lowering Feedwater Flow)"

Explanation (Optional): Procedure OP-27, Recirculation System has the operator place the SCOOP TUBE AUTO UNLOCK control switch in ON. This will allow a locked up reactor recirculation pump to run back if a FW pump is lost. If both pumps run back then the reactor will stabilize at a low power level.

Technical Reference(s):OP-27, "Reactor Recirculation System, SDLP-021, SDLP-33

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.09 †Recirculation flow mismatch: Plant-Specific 3.1 / 3.3

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis ___X__

10 CFR Part 55 Content: 55.41 __5___
55.43 __5___

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	203000 A1.04	
Importance Rating	_3.6_	_3.6_

Proposed Question: 46 / 41

Several minutes ago a medium break loss of coolant accident occurred simultaneously with a loss of off-site power. The reactor scrammed, core spray (CS) pumps, and the residual heat removal (RHR) pumps automatically started on a valid initiation signal. Reactor vessel level is at 125 inches and reactor pressure is at 500 psig, both level and pressure are dropping slowly due to the leak. The "B" Loop of RHR was in a normal standby lineup before the event and currently has the following indications.

"B" RHR pump ....OPERATING
"D" RHR pump..OPERATING
"B" Loop RHR Flow 10FI-133 (09-3 panel) ....0 gpm
"B" & "D" RHR pump discharge pressure.............200 psig
"B" Loop RHR minimum flow valve (MOV-16B)..OPEN
"B" Loop RHR injection valve (MOV-27B)............OPEN
"B" Loop RHR injection valve (MOV-25B)............CLOSED

Assuming that reactor level and pressure continue to drop how will the "B" Loop of RHR respond.

A. The "B" & "D" RHR pumps will not provide sufficient flow because they have been operating without minimum flow several minutes.

B. The "B" Loop RHR injection valve (MOV-25B) has failed to OPEN which will prevent the "B" Loop of RHR from injecting unless the injection valve is opened locally.

C. The RHR injection valve (MOV-25B) will OPEN when reactor pressure reaches 450 psig, and when reactor pressure drops below 200 psig indicated flow will rise and the minimum flow valve will close.

D. The "B" Loop RHR injection valve (MOV-25B) will OPEN and the minimum flow valve (MOV-16B) will CLOSE simultaneously when reactor pressure reaches 450 psig, indicated flow will rise as the minimum flow valve closes.

Proposed Answer: C. The RHR injection valve (MOV-25B) will OPEN when reactor pressure reaches 450 psig, and when reactor pressure drops below 200 psig indicated flow will rise and the minimum flow valve will close.

Explanation (Optional):

- The shutoff head of the pumps are approximately 200 psig. The pumps are running on minimum flow. The only indication that the pumps are on minimum flow is the position of the minimum flow valve.
- The MOV-25B is normally closed and will open on an initiation signal and reactor pressure less than 450 psig.
- The MOV-27B is OPEN in the standby line up.

- The minimum flow valve will close on flow of 1450 gpm, not a reactor pressure of 450 psig

Technical Reference(s): SDLP-10, RHR

Proposed references to be provided to applicants during examination: None

Learning Objective: A1. Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: (CFR: 41.5 / 45.5) A1.04 System pressure 3.6 / 3.6

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	206000 K6.03	
Importance Rating	_2.9_	_3.1_

Proposed Question: 47 / 42

A station blackout has occurred. What effect will the station blackout have on the HPCI system.

A. The HPCI system is not effected by a loss of AC power because there are no HPCI components that are powered by AC power.
B. The HPCI system will function; however, the flow controller will lose power and HPCI must be controlled locally.
C. The HPCI system will function; however, steam leakage along the shaft seal could occur.
D. The HPCI system will not start because the auxiliary oil pump will lose power and the turbine control valve will not open.

Proposed Answer: C. The HPCI system will function; however, steam leakage along the shaft seal could leak occur.

Explanation (Optional):
A. MOV 15 is powered from 600 VAC and will not automatically close on a HPCI isolation signal.
B. The HPCI flow controller will still have power it is powered from the "B" 125 VDC battery through an inverter.
D. The auxiliary oil pump is powered from DC.

Technical Reference(s): SDLP-23, HPCI

Proposed references to be provided to applicants during examination: None

Learning Objective: K6. Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM : (CFR: 41.7 / 45.7) K6.03 A.C. power: BWR-2,3,4 - 2.9 / 3.1*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content:
55.41 __7__
55.43 ______

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2_
Group #	__1__	__1__
K/A #	202002 K5.01	
Importance Rating	_2.8_	_2.8_

Proposed Question: 48 / 40

The plant is at power with both reactor recirculation (RWR) pumps operating at 90% speed. A significant oil leak occurs on the "A" RWR motor generator (MG) set piping. What effect will this have on the RWR system?

A. Both RWR pumps will automatically trip due to low oil pressure.
B. The "A" RWR pump scoop tube will lock up and the drive motor will trip on low oil pressure.
C. The "A" RWR pump scoop tube will lock up, allowing the oil in the fluid coupling reservoir to maintain the "A" RWR pump operating at a reduced speed.
D. The "A" RWR pump must be manually tripped.

Proposed Answer: B. The "A" RWR pump scoop tube will lock up and the drive motor will trip on low oil pressure.

Explanation (Optional): On low oil pressure the scoop tube will lockup and the drive motor breaker will trip, tripping the pump. The running pump flow will increase and sped must be reduced to less than 80% to prevent excessive jet pump differential pressure.

Technical Reference(s): SDLP-02H, AOP-8, Op-27

Proposed references to be provided to applicants during examination: None

Learning Objective: K5. Knowledge of the operational implications of the following concepts as they apply to RECIRCULATION FLOW CONTROL SYSTEM : (CFR: 41.5 / 45.3)
K5.01 Fluid coupling: BWR-3,4 - 2.8/2.8

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5___
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	209001 K4.01	
Importance Rating	_3.2_	_3.4_

Proposed Question: 49 / 43

Which one of the following statements correctly describes the automatic function of the core spray injection valves, 14MOV-11A(B) / 14MOV-12A(B)?

A. The core spray injection valves will open on a core spray automatic initiation signal when the core spray pump discharge pressure is 100 psig or greater to inject into the reactor vessel.

B. The core spray injection valves will open on a core spray automatic initiation signal when reactor pressure is less than 450 psig to protect the low pressure piping.

C. The core spray injection valves will automatically close at a reactor vessel level of 222.5 inches to prevent flooding and damaging the main steam lines.

D. The core spray injection valves will automatically close when the core spray sparger break detection logic is actuated to prevent flow diversion from the sparger.

Proposed Answer: B. The core spray injection valves will open on a core spray automatic initiation signal when reactor pressure is less than 450 psig to protect the low pressure piping.

Explanation (Optional): The core spray injections valves automatically open on an initiation signal and RPV pressure less than 450 psig. This protects the low pressure core spray piping. There are no automatic closures of the core spray injections valves on the RPV high level or sparger break detection logic.

Technical Reference(s): SDLP-14, CAD file S14-004.cdr

Proposed references to be provided to applicants during examination: None

Learning Objective: K4. Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: K4.01 Prevention of overpressurization of core spray piping 3.2 / 3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	217000 K3.02	
Importance Rating	_3.6_	_3.6_

Proposed Question: 50 / 50

A break has occurred in the steam supply line to the Reactor Core Isolation Cooling (RCIC) system upstream of the RCIC high flow sensing instrument taps (between the flow sensing taps and the reactor vessel). What effect will this have on RPV Pressure and RCIC?

A. RCIC will automatically isolate on high temperature in the drywell entrance area. The reactor will continue to depressurize.
B. RCIC will automatically isolate on high temperature in the drywell entrance area and stop the leak. Reactor pressure will be maintained by EHC and the turbine bypass valves.
C. RCIC will automatically isolate when sensed reactor pressure decreases to 50-100 psig. The reactor will continue to depressurize.
D. RCIC will automatically isolate and stop the leak when sensed reactor pressure decreases to 50-100 psig.

Proposed Answer: C. RCIC will automatically isolate when sensed reactor pressure decreases to 50-100 psig. The reactor will continue to depressurize.

Explanation (Optional): Based on SDLP-13, DWG# S13-012.cdr, a break upstream of the flow taps will not be isolated when a RCIC isolation occurs. In addition, the break is in the drywell and will not actuate the drywell entrance area temperature sensors. However, RCIC will isolate when reactor pressure decreases to 50-100 psig, but not isolate the leak.

Technical Reference(s): SDLP-13, LO 1.09

Proposed references to be provided to applicants during examination: None

Learning Objective: K3. Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: (CFR: 41.7 / 45.4) K3.02 Reactor vessel pressure 3.6 / 3.6

Question Source: Bank # INPO 7228
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Parent Question

QuestionId	7228	NSSSVendor	GE	CogLevel		Ka#	.217000.A2.15
		AbbrevLocName	Duane Arnold 1	ExamType		ILO	

Question:
A break has occurred in the steam supply line to the Reactor Core Isolation Cooling system upstream of the high flow sensing location. Which ONE of the following will provide system isolation antler this condition?

Reactor pressure low (50 psig)
RCIC emergency area cooler high temperature (175 deg F)
RCIC equipment room high vent inlet/outlet differential temperature (50 deg F)
Suppression pool area vent air high temperature (150 deg F)

Reference: ..217000.A2.15

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	211000 K2.01	
Importance Rating	_2.9_	_3.1_

Proposed Question: S51

The "B" standby liquid control (SLC) pump is tagged for maintenance. A fault has just occurred on MCC-152. What effect will this malfunction have on the current SLC limiting condition for operation (LCO).

A. Unaffected, no SLC components are powered from MCC-152.
B. Unaffected, this is the power supply for the "B" SLC pump motor which is already inoperable.
C. Both SLC subsystems are inoperable a more restrictive LCO must be entered.
D. Both SLC subsystems are inoperable however, the existing LCO is adequate.

Proposed Answer: C. Both SLC subsystems are inoperable a more restrictive LCO must be entered.

Explanation (Optional): MCC-152 powers the "A" SLC Pump and MCC-162 powers the "B" SLC pump. If both subsystems are inoperable the 7 day LCO must be shorted to a 24 hour LCO.

Technical Reference(s): SDLP-11, "SBL" , TS 3.4

Proposed references to be provided to applicants during examination: None

Learning Objective: K2. Knowledge of electrical power supplies to the following (CFR: 41.7) K2.01 SBLC pumps 2.9* / 3.1*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 __2__

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	212000 K1.15	
	Importance Rating	_3.8_	_3.9_

Proposed Question: 52 / 45

A reactor scram occurred 30 minutes ago. All plant systems responded as designed except for control rod 18-27. This control rod had a noticeable delay in starting to scram as compared to the other control rods. System engineering has performed a field inspection on 18-27 CRD hydraulic control unit (HCU) and determined that the "B" RPS scram pilot air valve did not reposition to scram the control rod. Based on this information, why did control rod 18-27 insert during the reactor scram?

A. Control rod 18-27 "A" RPS scram pilot air valve repositioned and bled the air off the scram outlet and inlet valves allowing them to open.
B. Control rod 18-27 "A" RPS scram pilot air valve repositioned and bled the air off the scram outlet valve allowing the valve to open and the rod to drift in.
C. Repositioning of scram pilot air valves on the other HCUs bled the scram air header down which resulted in control rod 18-27 scram outlet and inlet valves to open.
D. Repositioning of the backup scram valves bled the scram air header down allowing control rod 18-27 scram outlet and inlet valves to open.

Proposed Answer: D. Repositioning of the backup scram valves bled the scram air header down allowing control rod 18-27 scram outlet and inlet valves to open.

Explanation (Optional):

A. Both RPS scram pilot air valve must reposition to bled the air off the scram inlet and outlet valves.
B. Both RPS scram pilot air valve must reposition to bled the air off the scram inlet and / or outlet valves
C. Repositioning of the scram pilot air valve will block and prevent the scram air header from bleeding down.

Technical Reference(s): SDLP 05, Reactor Protection System

Proposed references to be provided to applicants during examination: None

Learning Objective: K1. Knowledge of the physical connections and/or cause- effect relationships between REACTOR PROTECTION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.15 SCRAM air header pressure 3.8/3.9

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X___
10 CFR Part 55 Content:	55.41 __2__	
	55.43 _____	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	__2__
	Group #	_____	__1__
	K/A #	215004 K2.01	
	Importance Rating	_2.6_	_2.8_

Proposed Question: S53

The plant is in a refueling outage and the core reload began six hours ago. The core is being reloaded in a Spiral pattern around the "A" source range monitor (SRM). Each SRM has two fuel bundles around it, except "A" which has 10, and all SRMs are indicating greater than 3 count per second (Attachment S53). There are no "dunking chambers" inserted into the core. Two minutes ago the System "B" 24/48 volt DC battery and chargers were removed from service for planned outage work. What effect does this have on the core reload?

A. The core reload may continue; however, fuel may only be moved in the core quadrants where the "A" & "B" SRMs are located.
B. The core reload may continue unrestricted because there are still two operable SRMs in the core.
C. The core reload must be stopped until the "A" SRM has its power restored.
D. The core reload must be stopped until the "B" or "D" SRM has its power restored.

Proposed Answer: D. The core reload must be stopped until the "B" or "D" SRM has its power restored.

Explanation (Optional): Loss of the "B" 24 VDC battery removed the "B" & "D" SRM from service. TS 3.10 B require that during core loading there are at least two SRMs operable. One in the quadrant where fuel is being moved and one in an adjacent quadrant. Since the core loading is around the "A" detector, the core loading must be stopped until power is restored to the "B" or "D" detector in an adjacent quadrant.

Technical Reference(s): SDLP-07B, TS 3.10

Proposed references to be provided to applicants during examination: Core Loading Map

Learning Objective: K2. Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 SRM channels/detectors 2.6/2.8

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41 __7__

55.43 __2__

Comments:

Examination Outline Cross-reference: Level		RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	215005 K3.07	
	Importance Rating	_3.2_	_3.3_

Proposed Question: 54 / 48

During a reactor startup the unit is at 35% power when the reactor operator selects the center control rod in preparation for withdrawal. The operator notices the following information on the 09-5 panel.

LPRM Status lights on the 4 Rod Display

Three DET A BYPASS lights lit
One DET B BYPASS light lit
Two DET C BYPASS lights lit
Zero DET D BYPASS lights lit

Based on this information what is the status of the RBM system.

A. The "A" RBM is automatically bypassed since there are too few inputs to the "A" RBM
B. The "B" RBM is automatically bypassed since there are too few inputs to the "B" RBM
C. The "A" RBM is providing a rod block to RMCS because there are too few inputs to the "A" RBM
D. The "B" RBM is providing a rod block to RMCS because there are too few inputs to the "B" RBM

Proposed Answer: C. The "A" RBM is providing a rod block to RMCS because there are too few inputs to the "A" RBM

Explanation (Optional): The RBM will provide a rod block signal to RMCS if there are to few LPRM inputs to the RBM circuitry. The RBM needs 50%, The "A" RBM does not have 50% of the LPRM inputs above the downscale values. The "A" uses the "A" and "C" level LPRMs and the "B" RBM uses the "B" and "D" LPRMs.

Technical Reference(s): SDLP-07C

Proposed references to be provided to applicants during examination: None

Learning Objective: K3. Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: (CFR: 41.7 / 45.4) K3.07 Rod block monitor 3.2/3.3

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 __7___	
	55.43 _____	

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1___
K/A #	216000 K4.14	
Importance Rating	_3.3_	_4.3_

Proposed Question: 55 / 49

The narrow range RPV level indication is calibrated for _(1)_ , the wide range is calibrated for _(2)_ and the fuel zone level indication is calibrated for _(3)_ .

A. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 1000 psig / 546°F and 135°F drywell temperature
(3) RPV 0 psig / 212°F and 212°F drywell temperature

B. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 0 psig / 212°F and 212°F drywell temperature
(3) RPV 0 psig / 212°F and 212°F drywell temperature

C. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 1000 psig / 546°F and 135°F drywell temperature
(3) RPV 1000 psig / 546°F and 135°F drywell temperature

D. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 1000 psig / 546°F and 135°F drywell temperature
(3) RPV 0 psig / 212°F and 135°F drywell temperature

Proposed Answer: A. (1) RPV 1000 psig / 546°F and 135°F drywell temperature
(2) RPV 1000 psig / 546°F and 135°F drywell temperature
(3) RPV 0 psig / 212°F and 212°F drywell temperature

Explanation (Optional): Fuel Zone is used under accident conditions and is therefore calibrated under accident conditions. The narrow range and wide range are calibrated under normal conditions.

Technical Reference(s): SDLP-02B

Proposed references to be provided to applicants during examination: None

Learning Objective: K4. Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.14 Temperature compensation for reactor water level indication: 3.3/ 3.4

Question Source: Bank # ________
Modified Bank # _______ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __7__	
	55.43 _____	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	217000 K5.07	
	Importance Rating	_3.1_	_3.1_

Proposed Question: 56 / 51

Reactor Core Isolation Cooling (RCIC) has initiated due to a low RPV water level. When RPV water level reaches 222.5", RCIC steam supply isolation valve 13MOV-131 closes.

RCIC will reinitiate when RPV water level lowers to less than:

A. 222.5 inches.

B. 126.5 inches.

C. 222.5 inches, BUT the RCIC turbine trip/ throttle valve must be locally reset.

D. 126.5inches, BUT the RCIC turbine trip/ throttle valve must be locally reset.

Proposed Answer: B. 126.5 inches

Explanation (Optional): RCIC Turbine Steam Inlet Isolation Valve 13MOV-131 will close when RPV water level reaches 222.5 inches, the trip throttle valve will not close. The "131" will stay closed until the RPV water level lowers to 126.5 inches at which time RCIC will auto-initiate.

Technical Reference(s):OP-19 REACTOR CORE ISOLATION COOLING SYSTEM

Proposed references to be provided to applicants during examination: None

Learning Objective: K5. Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) : (CFR: 41.5 / 45.3) K5.07 Assist core cooling 3.1 / 3.1

Question Source: Bank # FitzPatrick Requalification 21701003B01C Rev. 2
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __5__
55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	218000 K6.05	
	Importance Rating	_3.0__	_3.1__

Proposed Question:57 / 52

A loss of the 71ACUPS-2 relay room uninteruptable bus distribution panel has just occurred. What effect will the loss of this distribution panel have on the automatic depressurization system (ADS)?

A. The ADS "A" initiation logic channel will lose power and automatically swap to the "B" initiation logic channel.
B. ADS has lost power to the pilot valve solenoids and will only actuate mechanically on high reactor pressure in the relief mode.
C. Red/Green light indication will be lost on the 09-4 panel.
D. The "white" open indication light above each control switch will not illuminate when the valve is open because the valve monitoring system has lost power.

Proposed Answer: D. The "white" open indication light above each control switch will not illuminate when the valve is open because the valve monitoring system has lost power.

Explanation (Optional):

A loss of UPS power only effects the VMS system. Logic power is supplied from 125 VDC

Technical Reference(s):SDLP-02J "ADS," OP-68 "ADS".

Proposed references to be provided to applicants during examination: None

Learning Objective: K6. Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM : (CFR: 41.7 / 45.7) K6.05 A.C. power: 3.0*/ 3.1*

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __7___
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	223001 A1.11	
Importance Rating	_3.1_	_3.2_

Proposed Question: 58 / 53

A loss of coolant accident is in progress, torus and drywell sprays have been initiated. Which one of the following will result if these sprays are NOT terminated and torus pressure continues to drop below 0.0 psig?

A. Chugging at the outlet of the downcomer will result in structural failure of the downcomers.
B. The reactor building to torus vacuum breakers will open at differential pressure of 0.5 psid and partially de-inert the Primary Containment.
C. The torus downcomer ring header will fail (collapse) at a differential pressure of 0.5 psid between the torus and drywell.
D. The Reactor Building to Torus vacuum breakers will fail at differential pressure of 0.5 psid

Proposed Answer: B. The reactor building to torus vacuum breakers will open at differential pressure of 0.5 psid and partially de-inert the Primary Containment.

Explanation (Optional):
A. Chugging will not occur because the drywell sprays are on which will take the non condensibles back into the drywell.
C. The downcomer will not collapse at this pressure.
D. These vacuum breakers are design to open at this pressure.

Technical Reference(s): SDLP-16A

Proposed references to be provided to applicants during examination: None

Learning Objective: AI. Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES controls including: (CFR: 41.5 / 45.5) A1.11 Reactor building to suppression chamber differential pressure: Plant-Specific 3.1/ 3.2

Question Source: Bank # INPO 303

A LOCA is in progress and drywell sprays have been initiated. Which one of the following will result if drywell sprays are NOT terminated and drywell pressure lowers below 0.0 psig?

Partial de-inerting of the Primary Containment.

Chugging at the outlet of the downcomer.
Mechanical failure (collapse) of the Torus downcomer ring header.
Mechanical failure of the Reactor Building to Torus vacuum breakers

Reference: ..226001.K3.01

Phenomenon associated with initiation of DW sprays.
Phenomen associated with evaporative cooling due to spraying while in the unsafe region of the DW spray initiation limit curve.
Event is within the design of the vacuum breakers.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	223002 A2.06	
	Importance Rating	_3.0_	_3.2_

Proposed Question: 59 / 54

The plant is operating at 100% power with no other activities in progress. If the A1 primary containment isolation system (PCIS) condenser vacuum instrument were to fail to 0 inches mercury vacuum, what effect would this have on the plant and what operator actions would be necessary?

A. The inboard "A" main steam line isolation valve (MSIV) will close, the reactor will scram on high APRM flux and the other MSIVs will close on high steam flow. Implement EP-9, "Opening MSIVs" to reopen the MSIVs.
B. The outboard "A" main steam line isolation valve (MSIV) will close, the reactor will scram on high APRM flux and the other MSIVs will close on high steam flow. Implement EP-9, "Opening MSIVs" to reopen the MSIVs.
C. All MSIVs would remain OPEN. Manually scram the reactor and CLOSE all MSIVs.
D. All MSIVs would remain OPEN. Verify that an isolation is not required and verify that the appropriate TS actions are implemented.

Proposed Answer: D. All MSIVs remain OPEN. Verify that an isolation is not required and verify that the appropriate TS actions are implemented.

Explanation (Optional):
A. These conditions would not have resulted in closure of the "A" MSIV. The logic requires 1 out of 2 taken twice.
B. These conditions would not have resulted in closure of the "A" MSIV. The logic requires 1 out of 2 taken twice.
C. A single instrument failure will not result in an MSIV closure.
D. Correct. Verify that an isolation is not required and enter appropriate TS. ARP 09-5-1-55. PCIS SYSTEM A ISOLATION

Technical Reference(s): SDLP 16C, PCIS and ARP 09-5-1-55. PCIS SYSTEM A ISOLATION

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.06 Containment instrumentation failures 3.0/ 3.2

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 __5__	
	55.43 _____	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2___
	Group #	__1__	__1___
	K/A #	239002 A3.03	
	Importance Rating	_3.6_	_3.6_

Proposed Question: 60 / 55

Which of the following indications provide positive indication that a safety relief valve (SRV) is open.

A. Rise in the safety relief valve tail pipe temperature and white acoustic monitor light lit.
B. Safety relief valve red solenoid energized light lit and a rise in generator output.
C. Indicated rise in total core flow and white acoustic monitor light lit.
D. Indicated rise in total steam flow and drop in generator output.

Proposed Answer: A. Rise in the safety relief valve tail pipe temperature and white acoustic monitor light lit.

Explanation (Optional):
B. Generator output will decrease.
C. SRV open has no effect on total core flow (Comprehensive)
D. Indicated total steam flow decreases (Comprehensive)

Technical Reference(s): SDLP-71F and ARP-09-4-1-16

Proposed references to be provided to applicants during examination: None__

Learning Objective: A3. Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: (CFR: 41.7 / 45.7) A3.03 Tail pipe temperatures 3.6/3.6

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __7___
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	241000 A4.16	
Importance Rating	_3.3_	_3.2_

Proposed Question: 61 / 56

The plant is operating at 100% power when a transient occurs. You noticed a small upward spike on the APRMs and reactor pressure has taken a step rise of 3 psig and is constant at the new value. Reactor power remains at approximately 100%. Which one of the following indications, on the EHC console, is consistent with the event that has occurred.

A. "A" IN CONTROL light is ON
B. "B" IN CONTROL light is ON
C. LOAD LIMIT LIMITING light is ON
D. Mechanical Trip TRIPPED light is ON

Proposed Answer: B. EHC Console "B" IN CONTROL light is ON

Explanation (Optional): The indication of an APRM spike and then the step increase of 3 psig is the characteristic signature of a pressure regulator swap. The "A" pressure regulator is typically in control with the "B" pressure regulator set at 3 psig higher.

A. Since there has been no operator action to increase pressure and the pressure has increased by 3 psig the "A" in control light is OFF. It has failed in the upward direction.
C. This light is on when the load limiter is limiting the turbine load. This is set at 900 MWe (OP-9) if this were limiting load APRM would have increase significantly and stayed elevated.
D. If the mechanical trip light is ON the turbine has tripped. The turbine has not tripped.

Technical Reference(s):OP-9, AOP-6 and SDLP94C

Proposed references to be provided to applicants during examination: None

Learning Objective: A4. Ability to manually operate and/or monitor in the control room:(CFR: 41.7 / 45.5 to 45.8) A4.16 Lights and alarms 3.3/3.2

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__1__
	K/A #	259002 2.4.34	
	Importance Rating	_3.8_	_3.6_

Proposed Question: 62 / 58

A significant fire in the control room has resulted in the control room being evacuated. Which of the following indications should be used to obtain a value of reactor vessel water level from outside the control room in accordance with AOP-43, "Plant Shutdown From Outside the Control Room." (Actual reactor water level is 40 inches).

A. Wide range level indications on instrument rack 25-5
B. Narrow range level indications on instrument rack 25-5
C. Wide range level indications on instrument rack 25-6
D. Fuel Zone level indication on instrument rack 25-51

Proposed Answer: D. Fuel Zone level indication on instrument rack 25-51

Explanation (Optional): A&C. The wide range level indications can only be used for trend information and below 45 inches should not be used.
B. The Narrow range level indicator will be off scale for this level indication.

Technical Reference(s):AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.4.34 Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications. **RO 3.8/ SRO 3.6**

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 _____
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__1__
K/A #	226001 A4.07	
Importance Rating	_3.5_	_3.5_

Proposed Question: 63 / 60

The following plant conditions are present:

Torus pressure ....18.0 psig
RPV water level ....+30 inches indicated on the fuel zone level instruments
Drywell pressure and temperature ....Within the allowable region of the Drywell Spray Initiation Limit Curve

Which ONE of the following, by itself, will permit opening the Drywell spray valves for the 'A' RHR System?

A. Placing the DW & TORUS SPRAY VALVE OVERRIDE OF FUEL ZONE LEVEL Key lock Switch 10A-S18A in MANUAL OVERRIDE.
B. Closing the "A" LPCI Outboard Injection Valve 10MOV-27A.
C. Placing SPRAY CONTROL Switch 10A-S17A momentarily in RESET.
D. Placing SPRAY CONTROL Switch 10A-S17A momentarily in MANUAL.

Proposed Answer: D. Placing Containment Spray Control Switch 10A-S17A momentarily in MANUAL.

Explanation (Optional): The spray valves on the "A" loop will open if the following conditions are met:

1. Containment pressure >2.7 psig
2. SPRAY CONTROL switch is taken to the MAN position
3. RPV level is above 0 inches on the fuel zone instruments OR the DW & Torus Spray Valve Override Switch if taken to OVERRIDE.

A. This will not bypass any signal preventing valve opening if level is not less than TAF.
B. This will not allow the spray valves to be opened.
C. The switch must be taken to manual

Technical Reference(s):SDLP-10, pp53 and OP-13B

Proposed references to be provided to applicants during examination: None

Learning Objective: RHR / LPCI: Containment spray system mode A4 Ability to manually operate and/or monitor in the control room: (CFR 41.7/45.5 to 45.8) A4.07 Valve logic reset/ bypass/ override. 35. / 3.5

Question Source: Bank # FitzPatrick Requalification 0713.

The following plant conditions are present:

Torus pressure 18.0 psig
RPV water level +30 inches indicated by 02-3LI-91
Drywell pressure
and temperature Within the allowable region of the Drywell Spray Initiation Limit Curve

Which ONE of the following, by itself, will permit manual initiation of Drywell sprays from the 'A' RHR System?

E. Placing the DW & Torus Spray Valve Override of Fuel Zone Lvl Key lock Switch 10A-S18A in MANUAL OVERRIDE.
F. Closing the "A" LPCI Outboard Injection Valve 10MOV-27A.
G. Placing Containment Spray Control Switch 10A-S17A momentarily in RESET.
H. Placing Containment Spray Control Switch 10A-S17A momentarily in MANUAL.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__1__
K/A #	2620001 A3.02	
Importance Rating	_3.2_	_3.3_

Proposed Question: 64 / 76

The plant has just experienced a turbine trip / reactor scram and a fault on the 10400 bus. A residual transfer of electrical buses occurred. Which one of the following correctly describes the resultant configuration of the electrical buses?

A. Buses 10100, 10200, 10300, 10400 and 10700 de-energized
Buses 10300, 10400 10500 and 10600 energized from reserve power

B. Buses 10100 and 10200 de-energized;
Buses 10300, 10400 and 10700 energized from reserve power;
Buses 10500 and 10600 energized from either reserve power or EDGs

C. Buses 10400 and 10700 de-energized;
Buses 10100, 10200, 10300 energized from reserve power;
Bus10600 energized from either reserve power or EDGs
Bus10500 energized from EDGs

D. Buses 10100, 10200, 10400 and 10700 de-energized;
Bus 10300 energized from reserve power;
Bus10500 energized from either reserve power or EDGs
Bus10600 energized from EDGs

Proposed Answer: D. Buses 10100, 10200, 10400 and 10700 de-energized;
Bus 10300 energized from reserve power;
Bus10500 energized from either reserve power or EDGs
Bus10600 energized from EDGs

Explanation (Optional): Since Bus 10400 has faulted this bus will be dead and bus 10600 will be powered from the EDGs.

OP-22 Under normal operating conditions, the normal AC service power source and the off-site reserve AC power source are available to supply each emergency bus. The loss of the normal plant service power source results in automatic fast transfer to the off-site reserve AC power source. In the event of a failure to fast transfer, an automatic residual transfer will be made after a 3 second time delay, and each emergency diesel generator (EDG) will auto-start.

AOP-57 reserve supply breakers 10312 and 10412 close after bus voltage has been less than 25 percent of rate for greater than 3 seconds after the normal supply breaker trip.

Technical Reference(s): OP-46A, AOP-57, SDLP-71E

Proposed references to be provided to applicants during examination: None

Learning Objective:	A3. Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: (CFR: 41.7 / 45.7) A3.02 Automatic bus transfer 3.2/3.3 EO 1.03, 1.09, 1.14
Question Source:	Bank # LO331E~1.doc question 16 Modified Bank # A faulted bus 10400 was added to the question New ________
Question History:	Last NRC Exam ________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41	__7__
	55.43	_____

Examination Outline Cross-reference: Level

	RO	SRO	
Tier #	_____	__2__	
Group #		_____	__1__
K/A #		264000 A2.01	
Importance Rating		_3.5__ _3.6__	

Proposed Question: 65 / 61

The "A" EDG is paralleled to the 10500 bus for performance of ST-9BA, "EDG A and C Full Load Test and ESW Pump Operability Test." While preparing to parallel "C" EDG, a LOCA occurs coincident with a loss of off site power. What effect would this have on the plant and what operator actions would be necessary?

A. The "A" EDG has been declared inoperable for the surveillance. The second RHR pump will automatically start, no operator action required.

B. The 10500 bus will not separate from the 10300 bus. The "C" EDG will automatically load to bus 10500. Operator action will be required to start a second RHR pump.

C. The 10500 bus will separate from the 10300 bus. The "C" EDG will not automatically close in to the 10500 bus. Operator action will be required to start a second RHR pump.

D. The 10500 bus will separate from the 10300 bus. The "C" EDG will automatically close in to the 10500 bus. The second RHR pump will automatically start, no operator action required.

Proposed Answer: C. The 10500 bus will separate from the 10300 bus. The "C" EDG will not automatically close in to the 10500 bus. Operator action will be required to start a second RHR pump.

Explanation (Optional):

A. The plant configuration will result in a loss of load shedding capability on the 10500 bus and manual starting of RHR pump is required.
B. The "C" EDG will not load onto this bus because the "A" EDG will be able to maintain normal voltage at the bus.
D. The "C" EDG will not load onto this bus because the "A" EDG will be able to maintain normal voltage at the bus.

Technical Reference(s): ST-9BA, SDLP-93 (pp45, 46, 69), OP-22

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.01 Parallel operation of emergency generator 3.5/3.6

SDLP-93 1.05.b.1

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New __X__

Question History: Last NRC Exam ______

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __6__

55.43 ______

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__1__
K/A #	290001 A1.01	
Importance Rating	_3.1_	_3.1_

Proposed Question: 66 / 80

The unit is in a refueling outage with fuel moves in progress. The reactor building ventilation system is maintaining secondary containment at a negative 0.28 inches water differential pressure to atmosphere. The "A" standby gas treatment (SGT) train is operable and the "B" SGT train is tagged out of service for maintenance. Several minutes ago a fuel bundle was dropped over the core. The following indication are available in the control room.

Refueling Floor Exhaust
Radiation Monitor 17RM-456A ....1.2 x 10E 2 cpm

Refueling Floor Exhaust
Radiation Monitor 17RM-456B ....3.2 x 10E 5 cpm

Reactor Building Vent
Radiation Monitor 17RM-452A ....1.5 x 10E 2 cpm

Reactor Building Vent
Radiation Monitor 17RM-452B ....1.3 x 10E 2 cpm

Based on this information what is the expected configuration of secondary containment.

A. "A" SGT Train is ON, "B" SGT Train is OFF
Supply Isolation Valves 66AOV-100A & B CLOSED
Exhaust Isolation Valves 66AOV-101A & B CLOSED

B. "A" SGT Train is ON, "B" SGT Train is OFF
Supply Isolation Valve 66AOV-100A CLOSED
Exhaust Isolation Valve 66AOV-101A CLOSED

C. "A" & "B" SGT are OFF
Supply Isolation Valve 66AOV-100A CLOSED
Exhaust Isolation Valve 66AOV-101A CLOSED

D. "A" & "B" SGT are OFF
Supply Isolation Valve 66AOV-100B CLOSED
Exhaust Isolation Valve 66AOV-101B CLOSED

Proposed Answer: D. "A" & "B" SGT are OFF
Supply Isolation Vale 66AOV-100B CLOSED
Exhaust Isolation Valve 66AOV-101B CLOSED

Explanation (Optional): In this condition the 17RM-456B has detected the high radiation. The "A" has failed as is. Base on this information the "A" SGT system will not start and since

the “B” is out of service there will be no SGT running. The reactor building will have the “B” isolation valve close.

Technical Reference(s):LER 98001, ARP 09-75-1-15, OP-51A, OP-31pp 52

Proposed references to be provided to applicants during examination: None

Learning Objective: A1. Ability to predict and/or monitor changes in parameters associated with operating the SECONDARY CONTAINMENT controls including: (CFR: 41.5 / 45.5) A1.01 System lineups 3.1/3.1

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__1__	__2__
K/A #	201001 K2.05	
Importance Rating	_4.5_	_4.5_

Proposed Question: 67 / 37

The alternate rod insertion (ARI) valve solenoids 03SOV-201 through 205 are powered from which one of the following sources?

A. 71 ACUPS
B. 71AC-9
C. The "A" 125 Volt DC Battery / Battery Charger
D. The "B" 125 Volt DC Battery / Battery Charger

Proposed Answers: C. The "A" 125 VDC Battery / Battery Charger

Explanation (Optional): Power to these valves are supplied through 71DC-A5, CKT #7. This is a distribution panel off of the "A" 125 VDC battery / battery charger.

Technical Reference(s): SDLP-03C, Table III power supplies.

Proposed references to be provided to applicants during examination: None

Learning Objective: K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)
K2.05 Alternate rod insertion valve solenoids: 4.5*/4.5*

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level		RO	SRO
	Tier #	__2__	__2__
	Group #	__1__	__2__
	K/A #	201002 K3.01	
	Importance Rating	_3.4_	_3.4_

Proposed Question: 68 / 38

During weekly control rod drive testing the operator receives a reactor manual control timer malfunction. The timer malfunction can not be reset. How does the timer malfunction effect control rod movement?

A. Control rods can be inserted using the emergency in position of the emergency in/notch override switch.

B. Control rods can not be individually scrammed.

C. Control rods can not be selected to be moved because there is a select block

D. The rod worth minimizer blocks rod withdrawal because it loses rod position indication.

Proposed Answer: C. Control rods can not be selected to be moved because there is a select block

Explanation (Optional): Failure of the RMC timer will result in a select block. This will prevent a control rod from being selected and thus prevent any rod movement.

Technical Reference(s): SDLP-03F

Proposed references to be provided to applicants during examination: None

Learning Objective: K3. Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following: (CFR: 41.7 / 45.4)
K3.01 Ability to move control rods 3.4/3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	201006 K1.02	
Importance Rating	_3.4_	_3.4_

Proposed Question: 69 / 66

A startup is in progress at 30% power. In-sequence control rod 26-27 has just settled at its target out position of 24. Prior to selecting the next control rod, control rod 26-27 loses position indication at position 24. Which one of the statements below is an acceptable operator response to this condition.

A. No action is required because the rod worth minimizer (RWM) control rod blocks are disabled above the low power set point.

B. Bypass the RWM and insert control rod 26-27 to the RWM alternate position of 22.

C. No actions is required because the RWM will only apply insert blocks above the low power set point.

D. Insert a substitute position of 24 for control rod 26-27 and re-initialize the RWM to clear the rod block.

Proposed Answer: B. Bypass the RWM and insert control rod 26-27 to the RWM alternate position of 22.

Explanation (Optional): The loss of RPIS for a single control rod or the total loss of a RPIS will result in a RWM program abort. This will result in insert and withdrawal blocks from 0% power through 35% power. RWM rod blocks are automatically bypassed above the low power set point; however, program aborts are not. Since control withdraw is continuing AOP-26, Loss of RPIS will not allow the operator to insert a substitute position on a control rod.

Technical Reference(s): SDLP-03D, Rod Worth Minimizer, OP-64, Rod Worth Minimizer.

Proposed references to be provided to applicants during examination: None

Learning Objective: K1. Knowledge of the physical connections and/or cause- effect relationships between ROD WORTH MINIMIZER and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.02 RPIS: 3.4 / 3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis __x__

10 CFR Part 55 Content: 55.41 _2-9_
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	202001 K5.10	
Importance Rating	__2.8_	_2.8__

Proposed Question: 70 / 67

IF the speed controller for the first RWR pump MG Set to be placed in service was left at 50% prior to placing the MG set control switch to START, the MG set would:

A. Trip on overcurrent (50/51 relays).
B. Start and speed would rise to approximately 45%.
C. Overspeed because the generator field breaker would not close.
D. Start, speed would rise to 80%, then return to approximately 26%.

Proposed Answer: D. Start, speed would rise to 80%, then return to approximately 26%.

Explanation (Optional):

Technical Reference(s):See Above

Proposed references to be provided to applicants during examination: None

Learning Objective: K5. Knowledge of the operational implications of the following concepts as they apply to RECIRCULATION SYSTEM : (CFR: 41.5 / 45.3) K5.10 Motor generator set operation: Plant-Specific 2.8*/ 2.8

Question Source: Bank # FitzPatrick Requal No.20201004B03C Rev. 2

IF the speed controller for the first RWR pump MG Set to be placed in service was left at 50%, prior to placing the MG set control switch to START, the MG set would:

A. Trip on overcurrent (50/51 relays).
B. Start and speed would rise to approximately 45%.
C. Overspeed because the generator field breaker would not close.
D. Start, speed would rise to 80%, then return to approximately 26%.

Answer D

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __x___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __5___
55.43 _____

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	204000 K6.05	
Importance Rating	_2.6_	_2.6_

Proposed Question: 71 / 68

Maintenance has de-energized MCC-152 for planned maintenance. How does this maintenance effect the RWCU system?

A. The "A" RWCU pump (P-1A) will no longer have power available to the motor.
B. The "B" RWCU pump (P-1B) will no longer have power available to the motor.
C. The RWCU system outboard supply isolation valve (MOV-18) will lose power, and the supply line will have to be isolated in accordance with Technical Specifications.
D. The RWCU system inboard supply isolation valve (MOV-15) will lose power, and the supply line will have to be isolated in accordance with Technical Specifications.

Proposed Answer: D. The RWCU system supply isolation valve (MOV-15) will loose power, and the supply line will have to be isolated in accordance with Technical Specifications.

Explanation (Optional):
A. Supplied from MCC-131
B. Supplied from MCC-141
C. Supplied from "B" 125 VDC

Technical Reference(s): SDLP 12

Proposed references to be provided to applicants during examination: None

Learning Objective: K6. Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER CLEANUP SYSTEM : (CFR: 41.7 / 45.7) K6.05 A. C. power 2.6/2.6

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 __7___
55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__2__	__2__
	K/A #	214000 2.2.24	
	Importance Rating	_2.6_	_3.8_

Proposed Question: 72 / 70

The plant is at 100% power and maintenance has requested to remove the RPIS buffer card for control rod 06-31 which is currently at position 48. This will remove all position indication for this control rod. What actions, if any must be performed to release this maintenance activity?

A. A single rod scram time test must be performed.
B. A control rod coupling check must be performed.
C. The control rod must be declared inoperable.
D. No actions are necessary.

Proposed Answer: C. The control rod must be declared inoperable.

Explanation (Optional): A.&C. These tests are not required by TS for this activity
D. The control rod must be declared inoperable and electrically disarmed by TS 3.3.A.2b&d.

Technical Reference(s):AOP-26 LOSS OF ROD POSITION INDICATION*; TS 3.3.A.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.2.24 Ability to analyze the affect of maintenance activities on LCO status. RO 2.6/SRO 3.8

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _X__
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	215002 A2.03	
Importance Rating	_3.1_	_3.3_

Proposed Question: 73 / 71

The "C" average power range monitor (APRM) failed downscale, all other APRMs are indicating 100% power with no abnormal indications. What effect will this failure have on the RBM system and what actions must be taken to correct this condition.

A. The "A" RBM will generate a rod block. The "C" APRM and "A" RBM must be manually bypassed to clear the rod block.
B. The "A" RBM will generate a rod block. The "A" RBM must be manually bypassed to clear the rod block.
C. The "A" RBM will automatically be bypassed. The "C" APRM must be manually bypassed which places the "E" APRM in service as the "A" RBM reference APRM.
D. The "A" RBM will automatically be bypassed. The "C" APRM must be manually bypassed which places the "F" APRM in service as the "A" RBM reference APRM.

Proposed Answer: C. The "A" RBM will automatically be bypassed. The "C" APRM must be manually bypassed which places the "E" APRM in service as the "A" RBM reference APRM.

Explanation (Optional): A.& B. The RBM does not generate a rod block, it is automatically bypassed when the reference APRM is less than 30 power.
D. The correct reference APRM is E not F.

Note: The RBM will generate a rod block; however, since the reference APRM is downscale the RBM is automatically bypassed and the rod block is not passed to the RMCS.

Technical Reference(s):SDLP-07C

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.03 Loss of associated reference APRM channel: 3.1/3.3

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 __5___
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	245000 A3.06	
Importance Rating	_2.5_	_2.6_

Proposed Question: 74 / 74

The Unit has just experienced a turbine trip from 100% power and the turbine is currently costing down. Which one of the following describes the expected response of turbine support systems as the main turbine coasts down?

A. The Motor Suction pump and Emergency Bearing Oil pump will automatically start to supply the Main Turbine oil requirements.
B. The Motor Suction pump and Turning Gear Oil pump will automatically start to supply the Main Turbine oil requirements.
C. ONLY the Emergency Bearing Oil pump will start to supply the Main Turbine oil requirements.
D. NONE of the motor driven oil pumps will start.

Proposed Answer: B. The Motor Suction pump and Turning Gear Oil pump will automatically start to supply the Main Turbine oil requirements.

Explanation (Optional): SDLP-94A, states that during a turbine coast down the TGOP and MSP will automatically start. Both of these pumps start on low oil pressure.

Technical Reference(s):SDLP-94A pp. 52

Proposed references to be provided to applicants during examination: None

Learning Objective: A3. Ability to monitor automatic operations of the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS including:(CFR: 41.7 / 45.7) A3.06 Turbine lube oil pressure 2.5/2.6

Question Source: Bank # FitzPatrick Requal Exam Bank Question 0830
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

Level	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	262002 A4.01	
Importance Rating	_2.8_	_3.1_

Proposed Question: 75 / 77

The uninterruptible power supply (UPS) is being supplied by the alternate AC supply. Work has been completed and the UPS is ready to be powered from the preferred UPS source (motor generator set). What operator action, by procedure, must be taken in the control room to support transferring the UPS back to normal power?

A. Take manual control of both reactor feedwater pumps by lowering the MSC.
B. Lock up the "A" & "B" reactor water recirculation (RWR) scoop tubes.
C. Isolate the reactor water cleanup system to prevent an automatic isolation.
D. Drive the SRM and IRM detectors into the core.

Proposed Answer: B. Lock up the "A" & "B" reactor water recirculation (RWR) scoop tubes.

Explanation (Optional):
A. If there is a complete loss of the UPS <u>ONE</u> FW pump will be controlled using the MSC, not both.
B. This action is correct based on OP-27. All other actions are taken locally at the UPS panel.
C. This action is not required by procedure, only a verification of the isolation if a complete loss of the UPS occurs.
D. The SRMs and IRMs are not procedurally required to be driven into the core.

Technical Reference(s):AOP-21, OP-46b, OP-27 and SDLP-71F

Proposed references to be provided to applicants during examination: None

Learning Objective: A4. Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.01 Transfer from alternative source to preferred source 2.8/3.1

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	263000 2.4.10	
Importance Rating	_3.0_	_3.1_

Proposed Question: 76 / 78

The plant is operating at 100% power when annunciator 09-8-4-23, "EDG D CNTRL PWR LOSS," alarms. What effect does this condition have on the "D" emergency diesel generator (EDG) if a valid start signal were present?

A. 120 VAC UPS control power has been lost. The EDG will not auto-start.
B. 120 VAC UPS control power has been lost. The EDG will auto-start.
C. 125 VDC control power has been lost. The EDG will not auto-start.
D. 125 VDC control power has been lost. The EDG will auto-start.

Proposed Answer: C. 125 VDC control power has been lost. The "D" EDG will not auto-start.

Explanation (Optional): ARP 09-8-4-23, states that the loss of 125 VDC to the control circuit results in the loss of EDG D starting and shutdown capability.

Technical Reference(s): ARP 09-8-4-23

Proposed references to be provided to applicants during examination: None

Learning Objective: 263000 DC Electrical Distribution 2.4.10 Knowledge of annunciator response procedures. RO 3.0/ SRO 3.1

Question Source: Bank # INPO
Modified Bank # 7567

Loss of all 125 VDC power would have which one of the following effects on the Emergency Diesel Generators?

Answer Diesels would not start and could not be started at the Engine Control Panel

Diesels would auto-start and load.
Diesels would auto-start, but would not auto load.
Diesels would not auto-start but could be started at the Engine Control Panel.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	290003 A4.04	
Importance Rating	_2.8_	_3.0_

Proposed Question: 77 / 81

A loss of coolant accident has occurred. Secondary containment has isolated on high radiation in the reactor building vent exhaust. Annunciator 09-75-1-20, "CONTROL ROOM SUPPLY RADIATION MONITOR INOP OR HI," has just alarmed. What operator actions are required.

A. None. The control room HVAC system will isolate on a high control room supply radiation.
B. None. The control room HVAC System will isolate on a loss of coolant accident signal.
C. Place the control room ventilation isolation and purge control switch in PURGE to purge the control room area of any airborn radioactivity.
D. Place the control room ventilation isolation and purge control switch in ISOLATE to isolate the control room HVAC system.

Proposed Answer: D. Place the control room ventilation isolation and purge control switch in isolate to isolate the control room HVAC system.

Explanation (Optional): The control room HVAC does not have an automatic isolation function. On high radiation the operator must place the system in isolation mode. The use of purge mode is to remove smoke and stale air. It will increase the airborn activity in the control room if used under these circumstances.

Technical Reference(s):SDLP 70 and OP-55B, ARP 09-75-1-20

Proposed references to be provided to applicants during examination: None

Learning Objective: A4. Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.04 Environmental conditions 2.8/ 3.0

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

Level	RO	SRO
Tier #	__2__	__2__
Group #	__2__	__2__
K/A #	300000 A3.02	
Importance Rating	_2.9_	_2.7_

Proposed Question: 78 / 82

The Plant is operating at 100% power. The "A" air compressor is operating in LEAD, the "B" air compressor is operating in LAG and the "C" air compressor is in standby. After you received alarm AIR COMPRESSOR DISCHARGE TEMPERATURE HI on the "A" compressor, you verify the "A" second stage discharge air temperature is at 355°F and rising. As the temperature rises how will the instrument air system respond to this event? Assume no operator action.

A. The "A" and "B" air compressor will trip on high second stage discharge air temperature and when the air header pressure drops to 100 psig the "C" air compressor will automatically start.

B. The "A" air compressor will continue to run to failure because there is no high temperature trip on the second stage discharge air temperature. The "C" air compressor will automatically start when the breaker for the "A" air compressor opens.

C. The "A" air compressor will trip on high second stage discharge air temperature and when the air header pressure drops to 90 psig the "C" air compressor will start and maintain the air header pressure.

D. The "A" air compressor will trip on high second stage discharge air temperature and when the air header pressure drops to approximately 110 psig the "B" air compressor will load and maintain the air header pressure.

Proposed Answer: D. The "A" air compressor will trip on high second stage discharge air temperature and when the air header pressure decreases to approximately 110 psig the "B" air compressor will load and maintain the air header pressure.

Explanation (Optional): As the discharge air temperature on the "A" compressor increases it will trip at 360F. When the compressor trips the "B" which is operating in LAG will automatically load at 110 psig and maintain header pressure. The "C" which is in standby mode will not start because header pressure will not decrease to 100 psig.

A. Both running compressors will not trip on a high discharge temperature. Each compressor has its own temperature switch39TS-113A-C.

B. The "A" air compressor will trip at a temperature of 360.

C. The "C" compressor starts at 100 psig not 90 and the "B" will load and maintain the header pressure so that the "C" should not automatically start.

Technical Reference(s): SDLP-39, ARP 9-6-2-17, OP-39.

Proposed references to be provided to applicants during examination: None

Learning Objective: A3. Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including: (CFR: 41.7 / 45.7) A3.02 Air temperature 2.9/2.7

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __7__
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	2	2
Group #	2	2
K/A #	400000 A2.01	
Importance Rating	3.3	3.4

Proposed Question: 79 / 83

Lowering turbine building closed loop cooling (TBCLC) pressure will result in the standby pump starting at _(1)_ psig. If the TBCLC pressure can not be restored then _(2)_ .

A. (1) 75
(2) emergency service water will automatically start at 40 psig and inject into the TBCLC header.

B. (1) 75
(2) manually scram the reactor.

C. (1) 85
(2) emergency service water will automatically start at 40 psig and inject into the TBCLC header.

D. (1) 85
(2) manually scram the reactor.

Proposed Answer: D. (1) 85
(2) manually scram the reactor.

Explanation (Optional): The standby pump automatically starts at TBCLC discharge header pressure of 85 psig and if the standby pump fails to start then manually scram the reactor in accordance with AOP-47.

Technical Reference(s): AOP-47 Loss of TBCLC

Proposed references to be provided to applicants during examination: None

Learning Objective: A2. Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: (CFR: 41.5 / 45.6) A2.01 Loss of CCW pump 3.3/ 3.4

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

Question History: Last NRC Exam ______
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__3__	__3__
K/A #	215001 K6.04	
Importance Rating	_3.1_	_4.1_

Proposed Question: 80 / 84

During a full core LPRM calibration a Traversing In-Core Probe (TIP) detector becomes stuck in the core. The detector can not be moved in or out. If a loss of coolant accident were to occur with a valid containment isolation signal what operator action, if any must be taken to isolate the TIP system penetration?

A. No operator action is required. The TIP Shear valve automatically fires to cut the detector cable and seal the guide tube.
B. No operator action is required. The Guide tube ball valve automatically closes, to cut the detector cable and seal the guide tube.
C. Operator action is required. The TIP shear valve must be manually fired to cut the detector cable and seal the guide tube.
D. Operator action is required. The Guide tube ball valve must be manually closed to cut the detector cable and seal the guide tube.

Proposed Answer: C. Operator action is required. The TIP shear valve must be manually fired to cut the detector cable and seal the guide tube.

Explanation (Optional): The TIP ball valve does not have enough force to cut the detector cable. The TIP shear valve does not have an automatic function. Under these condition the TIP shear valve must be used because the detector is stuck and the penetration must be isolated.

Technical Reference(s): SDLP-07F, RAP-7.3.14

Proposed references to be provided to applicants during examination: None

Learning Objective: K6. Knowledge of the effect that a loss or malfunction of the following will have on the TRAVERSING IN-CORE PROBE: (CFR: 41.7 / 45.7)
K6.04 Primary containment isolation system: 3.1/3.4

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:
55.41 __7__
55.43 _____

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__2__	__2__
Group #	__3__	__3__
K/A #	233000 A1.03	
Importance Rating	_3.3_	_3.6_

Proposed Question: 81 / 85

The plant is maintaining 100% power. Several hours ago a leak on the fuel pool cooling pump discharge line resulted in the fuel pool cooling pumps tripping on low skimmer surge tank level. Maintenance expects to return the fuel pool cooling system to service in the next 5 hours. What temperature indications are available, under these conditions, to monitor fuel pool water temperature?

A. Fuel pool temperatures on the RHR & HPCI TEMP 10TRS-131 at the 09-21 panel.
B. Observing local annunciator alarm lights at fuel pool filter demineralizer panel
C. Running the residual heat removal system in fuel pool cooling assist and monitoring RHR heat exchanger temperatures.
D. Installing a temporary temperature indication in the spent fuel pool.

Proposed Answer: D. Installing a temporary temperature indication in the spent fuel pool.

Explanation (Optional):
A. This chart recorder only has fuel pool HX inlet and outlet as well as pump suction temperatures. The water in this area is no longer communicating with the fuel pool.
B. This local panel does not have any temperature alarms on it.
C. To use RHR the plant must be in cold shutdown. There is not enough time to get the plant to a cold shutdown condition before fuel pool cooling is expected to be returned to service.
D. Procedure AOP-68 FUEL POOL COOLING AND CLEANUP TROUBLE* has the operator install a temperature indication in the fuel pool as need for this type of event.

Technical Reference(s): AOP-68 FUEL POOL COOLING AND CLEANUP TROUBLE

Proposed references to be provided to applicants during examination: None

Learning Objective: AI. Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: (CFR: 41.5 / 45.5) AI.03 Pool temperature 3.1/ 3.3

Question Source: New X

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X___

10 CFR Part 55 Content: 55.41 __5__
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__2__	__2__
	Group #	__2__	__3__
	K/A #	239001 A2.07	
	Importance Rating	_3.8_	_3.9_

Proposed Question: 82 / 73

The plant is operating at 85% power when a transient results in fuel failure and a small main steam line (MSL) leak in the steam tunnel. What is the expected plant response and what operator actions will be required after the automatic actions have completed?

A. High MSL flow will result in a reactor scram and high MSL temperature will close the main steam isolation valves (MSIVs). Implement actions in AOP-1, Scram and EOP-2, RPV control.

B. High MSL radiation will result in a reactor scram and high steam tunnel temperature will close the MSIVs. Implement actions in AOP-1, Scram and EOP-4, Primary Containment Control.

C. High steam tunnel temperature will close the MSIVs and the reactor will scram on the MSIV valve position. Implement actions in AOP-1, Scram and EOP-2, RPV control.

D. High steam tunnel temperature will close the MSIVs and the reactor will scram on high reactor vessel pressure. Implement actions in AOP-1, Scram and EOP-4, Primary Containment Control .

Proposed Answer: C. High steam tunnel temperature will close the MSIVs and the reactor will scram on the MSIV valve position. Implement actions in AOP-1, Scram and EOP-2, RPV control.

Explanation (Optional):
A. High MSL flow does not result in a scram.
B. The MSL high radiation scram / isolation has been disabled.
C. Correct.
D. The reactor will not scram on high reactor vessel pressure. The reactor will scram on MSIV valve position.

The MSIV will close from high steam tunnel temperature and the closure of the MSIVs will result in a scram and high vessel pressure. Primary containment control will be needed as soon as the MSIVs close due to the SRV lifting.

Technical Reference(s): SDLP-16C

Proposed references to be provided to applicants during examination: EOPs

Learning Objective: A2. Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.07 Main steam area high temperature or differential temperature high 3.8/3.9

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

Question History: Last NRC Exam ______

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	__5__
	55.43	_____

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__2__
Group #	____	__3__
K/A #	290002 2.1.11	
Importance Rating	_3.0_	_3.8_

Proposed Question: S83

A 3D Monicore Periodic Log has just printed out. The following is a summary of information that is listed on the log.

POWER MWT 2530
POWER MWE 855
FLOW MLB/HR 75.453
MFLCPR 0.938 37-24
MFLPD 1.020 41-24-6
MAPRAT 0.919 39-22-4
PCRAT 1.004 15-32-4
LOAD LINE 101.3%

Based on this information what is the required actions by Technical Specification?

A. MFLPD is greater than 1.0, action must be taken within 15 minutes to reduce MFLPD less than 1.0 and if not corrected in 2 hours be less than 25% power in the next 4 hours.
B. MFLCPR is less than 1.0, action must be taken within 15 minutes to increase MFLCPR greater than 1.0 and if not corrected in 2 hours be less than 25% power in the next 4 hours.
C. PCRAT is greater than 1.0, action must be taken within 15 minutes to decrease PCRAT less than 1.0 and if not corrected in 2 hours be less than 25% power in the next 4 hours.
D. Reactor thermal power has exceed the licensed thermal power limit, action must be taken within the next hour to reduce the shift average thermal power below the licensed thermal power limit.

Proposed Answer: A. MFLPD is greater than 1.0, action must be taken within 15 minutes to reduce MFLPD less than 1.0 and if not corrected in 2 hours be less than 25% power in the next 4 hours.

Explanation (Optional):
A. TS 3.5.I
B. MFLCPR of less than 1.0 is in compliance with Technical Specifications.
C. PCRAT is not in Technical Specifications. This is a measure of how close the pin power is to the preconditioned threshold and may be above 1.0.
D. 2530 MWT is less than the licensed thermal power limit in the license.

Technical Reference(s): Technical Specification 3.5.I, SDLP-09B

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.1.11 Knowledge of less than one hour technical specification action statements for systems. RO 3.0/SRO 3.8

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

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	__X__

10 CFR Part 55 Content:	55.41	_____
	55.43	__2__

Comments:

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	____	__3__
	Group #	____	____
	K/A #	2.1.6	
	Importance Rating	____	_4.3__

Proposed Question: S84

The shift manager was notified via phone that the "A" reactor water recirculation (RWR) pump vibration annunciator alarmed and drywell pressure was rising. Two minutes later when the shift manager enters the control room he observes that the reactor has scrammed on high drywell pressure and the crew is carrying out the following actions.

The control room supervisor (CRS) and senior nuclear operator (SNO) are at the 09-4 panel trying to isolate the "A" RWR pump.

The nuclear control operator 1 (NCO1) is at the 09-3 panel performing manipulations on the high pressure coolant injection system.

The NCO2 is on the phone with radiation protection.

The controller is directing CRS & SNO actions using EOP-2, "RPV Control" and EOP-4, "Primary Containment Control."

Given these conditions what actions must the shift manager take?

A. Maintain oversight of the control room and provide direction to the controller as he directs implementation of the EOPs

B. Maintain oversight of the control room and direct the CRS to implement the EOPs and have the controller perform independent assessor duties.

C. Implement the EOPs and have the NCO2 full fill the emergency director position.

D. Implement the EOPs and fulfill the emergency director position.

Proposed Answer: B. Maintain oversight of the control room and direct the CRS to implement the EOPs and have the controller perform independent assessor duties.

Explanation (Optional):

A. The controller duties do not involve line supervision of operators and he is also required to be independent of the control room staff.
C. The NCO2 is not a qualified emergency director.
D. The SM should not direct the implementation of the EOPs, he/she should perform oversight and STA duties.

Technical Reference(s): AP-12.03 Administration of Operations

Proposed references to be provided to applicants during examination: None

Learning Objective:	2.1.6, Ability to supervisor and assume a management role during plant transients and upset conditions. EO 46.02 and 46.03j,	
Question Source:	Bank #	______
	Modified Bank #	______ (Note changes or attach parent)
	New	__X__
Question History:	Last NRC Exam	______

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41	____
	55.43	__5__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__3__	__3__
Group #	____	____
K/A #	2.1.10	
Importance Rating	_2.7_	_3.9_

Proposed Question: 85 / 88

The plant has been at a steady state power for the last 7 days during a feedwater flow calibration. The calibration is complete and the final adjustments have been made to the feedwater flow instrument loop. The first 3D Monicore output after these adjustments were made has the following information:

POWER MWT 2580
POWER MWE 860
FLOW MLB/HR 75.453
MFLCPR 0.938 37-24
MFLPD 0.946 41-24-6
PCRAT 1.004 15-32-4
PR (PSIa) 1057
LOAD LINE 101.7%

Based on this information what actions must the operator take?

A. Reduce reactor pressure to provide more margin to the high pressure reactor scram.
B. Thermal power must be reduced to 2536 MWt or less within 15 minutes.
C. Contact reactor engineering within 15 minutes to decrease PCRAT less than 1.0.
D. The reactor must be scrammed.

Proposed Answer: B. Thermal power must be decrease to 2536 MWt or less within 15 minutes.

Explanation (Optional):
A. MFLCPR is less than 1.0 as required by TS
B. Correct - RAP 7.3.16 requires the plant to reduce power to less than 2536 MWt within 15 minutes.
C. There is no need to reduce PCRAT to less than 1.0, because the plant has been at this power level for 7 days. The envelop must be updated.
D. There is no reason to scram the reactor.

Technical Reference(s): RAP 7.3.16

Proposed references to be provided to applicants during examination: None

Learning Objective: Knowledge of conditions and limitations in the facility license. RO 2.7 / SRO 3.9

LO NET 238.3, 1.03.j Rated Thermal Power

Question Source: Bank # FitzPatrick Requal 33301036COLRS01 Rev.1

The plant is at full power when it is discovered that actual thermal power is 2580 MWt.

Select the following required action:

A. Scram the reactor.

B. Reduce power to within 2536 MWt within 15 minutes.

C. Commence an orderly shutdown by inserting the RSCS groups per RAP-7.3.16.

D. Quickly insert the cram rods per RAP-7.3.16 and notify the Nuclear Engineer.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _5, 10
55.43 _____

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	_____	__3__
Group #	_____	_____
K/A #	2.1.34	
Importance Rating	_2.3_	_2.9_

Proposed Question: S86

At the time of a reactor startup reactor coolant water had the following analysis:

Reactor Water Chloride ion 0.1 ppm
Reactor Water Conductivity 3 micromho/cm

Twelve hours after placing the reactor in the power operating condition the levels have changed to:

Reactor Water Chloride ion 0.08 ppm
Reactor Water Conductivity 6 microhmo/cm

Which one of the following requirements will apply?

A. The reactor must be placed in cold condition within 24 hours with water quality brought within limits for shutdown.

B. Operation may continue unless chemistry continues to degrade or steaming rate drops below 100,000 lb/hr.

C. Unless conductivity improves in the next 12 hours, the plant must be placed in cold condition within the following 24 hours.

D. Unless chloride ion concentration improves in the next 12 hours, the plant must be placed in cold condition within the following 24 hours.

Proposed Answer: C. Unless conductivity improves in the next 12 hours, the plant must be placed in cold condition within the following 24 hours.

Explanation (Optional): Reactor water chloride is within limits. The reactor water conductivity is not and has not been, however, it is within the limits of 10 microhmo/cm (1.1.A.2) and has 12 hours left to get within 5 microhoms/cm or be in cold shutdown in the following 24 hours.

Technical Reference(s): AP-01.04 TECH SPEC RELATED REQUIREMENTS, LISTS, AND TABLES* Section 5

Proposed references to be provided to applicants during examination: AP-01.04 TECH SPEC RELATED REQUIREMENTS, LISTS, AND TABLES* Section 5

Learning Objective: 2.1.34 Ability to maintain primary and secondary plant chemistry within allowable limits. RO 2.3 / SRO 2.9

Question Source:
Bank # FitzPatrick bank 33301037RPVS01 Rev. 0
Modified Bank # ________ (Note changes or attach parent)
New ________

The following changes have been made to the question to improve the clarity:

Twelve hours after reaching full power the levels have changed to:

was changed to

Twelve hours after placing the reactor in the power operating condition the levels have changed to:

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:
55.41 _____
55.43 __5__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__3__
Group #	____	____
K/A #	2.2.19	
Importance Rating	____	_3.1_

Proposed Question: S87

The "A" residual heat removal (RHR) pump and associate equipment has just been taken out of service for a 2 day planned maintenance LCO window. The following tasks are listed as minor maintenance.

Change chart recorder pens for RHR flow recorder.
Obtain an oil sample from the "A" RHR motor.
Build scaffolding around the "A" RHR pump.
Replace 2 damaged bolts on the RHR pump flange.
Repairing a Gai-tronics phone near the RHR pump.

Can this work be released to be performed as minor maintenance?

A. Yes all work listed can be performed as minor maintenance.
B. No repair of the Gai-tronics phone can not be performed under minor maintenance.
C. No replacement of the 2 damaged bolts on the RHR pump flange can not be performed under minor maintenance.
D. No building scaffolding around the RHR pump can not be performed under minor maintenance.

Proposed Answer: C. No replacement of the 2 damaged bolts on the RHR pump flange can not be performed under minor maintenance.

Explanation (Optional): AP-10.01 states that Minor Maintenance can not be used for work on Cat I or M components that require parts/materials replacement. These shall be identified on a PID. In addition this should be an ISI code class 3 boundary greater than 1 inch..

Technical Reference(s): AP10.01pp 70

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.2.19 Knowledge of maintenance work order requirements.

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _X_
Comprehension or Analysis ___

10 CFR Part 55 Content:	55.41	_____
	55.43	__5__
	55.45	_13__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__3__
Group #	____	____
K/A #	2.2.21	
Importance Rating	____	_3.5_

Proposed Question: S88

During the performance of ST-4E, "HPCI AND SGT LOGIC SYSTEM FUNCTIONAL AND SIMULATED AUTOMATIC ACTUATION TEST*," it was noted that turbine vibration on 23 VM-100 read 0.365 IPS (High Limit of 0.385 IPS). The local reading was 1.92 mils (High Limit of 2.0 mils) using an IRD-810. A review of past completed surveillance tests indicates steady performance over the last several years of 0.100 IPS on the VM-100 and 1.00 mils on the IRD-810. The ST Acceptance Criteria were all met. Instrumentation calibration was verified to be current. Based upon this information, HPCI should be considered to be in a(n)...

A. Technical Specification Inoperable Condition

B. Degraded Condition

C. Non-conforming Condition

D. Maintenance Rule Unavailable Condition

Proposed Answer: B. Degraded Condition

Explanation (Optional)

A. The system has not yet demonstrated itself as inoperable or incapable of performing its intended function.

B. The system is degraded in that a loss of quality or function capability has been experienced as indicated by a noticeable increase in parameters that are precursors to failure (vibration).

C. The conditions do not describe a deficiency in characteristic, documentation or procedure that renders the quality of the vibration measurements unacceptable or indeterminate.

D. The system is not inoperable therefore it si not Unavailable. The system is still capable of performing its intended function.

Technical Reference(s): AP-03.11 pp 8-10

Proposed references to be provided to applicants during examination: None

Learning Objective: Knowledge of pre and post maintenance operability requirements. RO 2.3 / SRO 3.5

LPAD, Objectives 13.02.a, 10.02.b, and 17.02.g
NET-238.3, Objective 1.03.h

Question Source: NEW

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	_____
	55.43	_2, 3_

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__3__
Group #	____	____
K/A #	2.2.20	
Importance Rating	____	_3.3_

Proposed Question: S89

Several hours ago an event occurred. The plant is still at power and a troubleshooting plan has been developed. Which one of the troubleshooting items listed below must be controlled as a temporary modification for troubleshooting.

A. Installing a drain hose to support flushing an out of service reactor feedwater pump turbine lube oil cooler.
B. Installing a chart recorder on the feedwater level control system instrument loop.
C. Installation of jumpers in accordance with a surveillance procedure.
D. Changing the feedwater level column selector switch between the "A" and "B" position.

Proposed Answer: B. Installing a chart recorder on the feedwater level control system instrument loop.

Explanation (Optional):
A. The Cooler is out of service and is therefore excluded via AP-05.02, step 2.5
C. Actions performed to facilitate an instrument calibration or test, provided the system is restored immediately following the activity does not require a temporary modification.
D. By moving the switch this does not alter the design of the feedwater system. OP-2A has you maintain the switch in either position.

Technical Reference(s):OP-2A FEEDWATER SYSTEM

Proposed references to be provided to applicants during examination: None

Learning Objective: Knowledge of the process for managing troubleshooting activities. RO 2.2 / SRO 3.3

Question Source: Bank # ______
Modified Bank # INPO 2539

WHICH ONE (1) of the following is a Temporary Modification?

Connecting cables from a 480v Motor Control Center (MCC) to a temporary power panel for outage maintenance support.

Performing a channel calibration procedure, which requires installing jumpers to electrically bypass automatic actuation.

A blank flange is installed on a line while rerouting the line under an approved Work Order.

Maintenance technicians installing a temporary drain hose to support changing oil in a pump.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 __3__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__3__
Group #	____	____
K/A #	2.2.25	
Importance Rating	____	_3.7_

Proposed Question: S90

What is the Technical Specification basis for the turbine stop valve closure scram signal when above 29% rated power?

A. This scram protects the minimum critical power ratio safety limit by anticipating the neutron flux, heat flux and pressure increase due to a load rejection which exceeds the bypass valve capability.
B. This scram protects the minimum critical power ratio safety limit by anticipating the neutron flux, heat flux and pressure increase due to a rapid closure of these valves without turbine bypass valves available.
C. This scram protects the reactor vessel high pressure safety limit by anticipating the pressure increase due to a rapid closure of these valves.
D. This scram protects the fuel from exceeding the linear heat generation rate safety limit by anticipating the neutron flux due to the rapid valve closure.

Proposed Answer: B. This scram protects the minimum critical power ratio safety limit by anticipating the neutron flux, heat flux and pressure increase due to a rapid closure of these valves without turbine bypass valves available.

Explanation (Optional):
A. The low EHC header pressure protects against a load reject.
B. Correct
C. Closure of the stop valves will not exceed the RPV safety limit. The MSIV closure event with out scram will have a larger effect on the safety limit because of the decreased volume between the MSIV and RPV.
D. This scram does help limit the heat flux and LHGR, however the APRM provide the protection against LHGR, not this scram. In addition, this is not discussed in the basis and this is not a safety limit.

Technical Reference(s): Technical Specifications page 19

Proposed references to be provided to applicants during examination: None

Learning Objective: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____
55.43 __2__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__3__
Group #	____	____
K/A #	2.2.32	
Importance Rating	____	_3.3_

Proposed Question: S91

Which one of the following activities would constitute a core alteration as defined in Technical Specification.

A. Withdrawal of source range monitors (SRM) to verify proper signal to noise ratio.
B. Withdrawal of a control rod from position 00 to 48.
C. Replacement of a local power range monitor (LPRM) string.
D. Running traversing in-core probes through LPRM strings to verify proper operation.

Proposed Answer: C. Replacement of a local power range monitor (LPRM) string.

Explanation (Optional): All other answers can be accomplished by moving the components via their normal means. The only item that is not performed under its normal means is the removal of the LPRM string.

TS 1.B Core Alteration The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

Technical Reference(s):TS 1.B

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.2.32 Knowledge of the effects of alterations on core configuration.

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

Question History: Last NRC Exam __________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X___
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 _2, 6_

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	____	__3__
	Group #	____	____
	K/A #	2.3.3	
	Importance Rating	____	_2.9_

Proposed Question: S92

A canal discharge has been ordered. To start the discharge the Auxiliary Operator must obtain the key to open the canal flow control valves. Which one of the individuals listed below controls this key?

A. The shift manager
B. The Radwaste Supervisor
C. Control Room Supervisor
D. The Security Supervisor

Explanation (Optional): A. The shift manager

Technical Reference(s):OP-49,LIQUID RADIOACTIVE WASTE SYSTEM*

Proposed references to be provided to applicants during examination: None

Learning Objective: Knowledge of SRO responsibilities for auxiliary systems that are outside the control room (e.g. / waste disposal and handling systems). RO 1.8 / SRO 2.9

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __x__
Comprehension or Analysis ____

10 CFR Part 55 Content:
55.41 ____
55.43 __4__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__3__
Group #	____	____
K/A #	2.3.6	
Importance Rating	____	_3.1_

Proposed Question: S93

A liquid radioactive waste discharge is required to reduce the waste sample tank level. The radwaste effluent radiation monitor is out of service for maintenance. Who must approve the discharge permit and what special actions, if any, must be taken to start the discharge?

A. The radwaste supervisor must approve the discharge. The radwaste effluent monitor must be returned to service before the discharge can be started.
B. The general manager - plant operations must approve the discharge permit. Discharge flow must be estimated once per 4 hours during the discharge.
C. The shift manger must approve the discharge permit. The radwaste effluent monitor must be returned to service before the discharge can be started.
D. The shift manager must approve the discharge permit. Two technically qualified members of the facility staff must verify the discharge line valving before the discharge can be started.

Proposed Answer: D. The shift manager must approve the discharge permit. Two technically qualified members of the facility staff must verify the discharge line valving before the discharge can be started.

Explanation (Optional): A.&C. The radwaste effluent monitor does not have to be returned to service before the discharge starts.
B. The general manger - operations is not required to approve the discharge permit and the flow rate does not have to be estimated every four hours.

Technical Reference(s):OP-49, TS Table 2.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.3.6 Knowledge of the requirements for reviewing and approving release permits.

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____

55.43 __2, 5__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	____	__3__
Group #	____	____
K/A #	2.3.10	
Importance Rating	____	_3.3_

Proposed Question: S94

Power had been reduced to 60% and Hydrogen injection was shutdown, in accordance with a radiation work permit, to perform maintenance activities on a moisture separator drain valve. The following dose rates were obtained and used to plan the work activities.

100% power & Hydrogen injection OPERATING...............1100 mrem/hr
60 % power & Hydrogen injection OPERATING...............800 mrem/hr
60 % power & Hydrogen injection SHUTDOWN...............200 mrem/hr

The work was completed several hours ago, the unit is at 100% power and Hydrogen injection is operating. On a plant tour you notice that the door to the condenser bay is blocked open and there are 2 people entering the condenser bay. They state that they are on the same radiation work permit that was used to perform work on the moisture separator drain valve and they were told to clean up the area. What actions if any, must be taken?

A. The activity may continue because they are on the radiation work permit that was used to perform the repair work on the moisture separator drain valve.
B. The activity may continue because the workers have electronic dosimeters that will alarm and the workers will leave the area if the general area dose rates are significantly higher than expected.
C. The activity must be stopped and the condenser bay door closed, locked and posted high radiation area.
D. The activity must be stopped and the condenser bay door closed, locked and posted very high radiation area.

Proposed Answer: C. The activity must be stopped and the condenser bay door closed, locked and posted high radiation area.

Explanation (Optional): The unit has returned to power. Dose rates in the area of the moisture separator have returned to 1100 mrem/hr. The door should be locked and controlled as a high rad area. The personnel RWP are not valid for this condition because power is at 100% and Hydrogen injection is operating.

Technical Reference(s):AP-07.06 HIGH RADIATION AREA CONTROL, TS 6.11

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.3.11, Ability to perform procedure to reduce excessive levels of radiation and guard against personnel exposure.

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____
55.43 __4__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__3__	__3__
Group #	_____	_____
K/A #	2.3.11	
Importance Rating	_2.7_	_3.2_

Proposed Question: 95 / 96

A main steam line break outside of containment has occurred. The main steam line isolation valves have successfully isolated the break and the reactor scrammed. What actions can be taken to reduce radiation leakage to the environment in accordance with AOP-40 MAIN STEAM LINE BREAK.

A. Dispatch a team to verify reactor building integrity and start the main steam leakage collection system from the control room.
B. Dispatch a team to verify turbine building integrity and start the main steam leakage collection system from the relay room.
C. Dispatch a team to verify turbine building integrity and start the main steam leakage collection system from the remote shutdown panels.
D. Perform an emergency reactor depressurization.

Proposed Answer: B. Dispatch a team to verify turbine building integrity and start the main steam leakage collection system from the relay room.

Explanation (Optional): A. The MSLC system is manually operated from the relay room and the procedure directs teams to be dispatched to the turbine building. An RPV emergency depressurization is not required because there is no primary system discharging into secondary containment.

Technical Reference(s):AOP-40 MAIN STEAM LINE BREAK

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.3.11, Ability to control radiation releases.

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __10_
55.43 _____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	__3__	__3__
	Group #	_____	_____
	K/A #	2.4.1	
	Importance Rating	_4.3_	_4.6_

Proposed Question: 96 / 97

The plant was operating at 100 % power when an event occurs. The following plant conditions are present:

APRM Reactor Power	15%	
Reactor Level	200 inches	
Reactor Pressure	1145 psig	
Drywell Temperature	135°F	
Drywell Pressure	2.9 psig	
Torus Water Temperature	95°F	
Torus Pressure	1.0 psig	
Torus Level	13.9 Feet	
Reactor Building to outside dP	- 0.29 inches water	
Reactor Building Temperatures	RWCU heat exchanger room	105°F
	RWCU "A" Pump Room	110°F
	RWCU "B" Pump Room	110°F
Reactor Building Radiation Levels	RWCU Heat Exchanger Room	30 mr/hr
	RWCU Pump Area	20 mr/hr

Based on these conditions what emergency operating procedures should be entered/used and what operator actions have been taken in the first minute?

A. EOP-2, RPV Control; EOP-3 Failure to Scram; and EOP-4, Primary Containment Control have been entered and drywell sprays have been initiated.
B. EOP-2, RPV Control; EOP-3 Failure to Scram; and EOP-4, Primary Containment Control have been entered and a manual scram initiated and mode switch placed in shutdown.
C. EOP-3 Failure to Scram; EOP-4, Primary Containment Control; and EOP-5 Secondary Containment Control have been entered and the mode switch has been taken to shutdown.
D. EOP-3 Failure to Scram; EOP-4, Primary Containment Control; and EOP-4a, Primary Containment Gas Control have been entered and SRMs & IRMs are driving into the core.

Proposed Answer: B. EOP-2, RPV Control; EOP-3 Failure to Scram; and EOP-4, Primary Containment Control have been entered and a manual scram initiated and mode switch placed in shutdown.

Explanation (Optional):
A. There are no conditions that would require drywell sprays to be initiated.
C. There are no entry condition for EOP-5.
D. There is adequate core cooling, therefore, there is no entry conditions for EOP-4a.

Technical Reference(s): FSAR Section 14, EOPs

Proposed references to be provided to applicants during examination: EOPs w/o entry conditions.

Learning Objective: 2.4.1 Knowledge of EOP entry conditions and immediate action steps.

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

Question History: Last NRC Exam ________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __10__
55.43 __5__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__3__	__3__
Group #	_____	_____
K/A #	2.4.18	
Importance Rating	_2.7_	_3.6_

Proposed Question: 97 / 98

In EOP-4, "Primary Containment Control," the primary containment pressure section (PC/P) requires that torus pressure exceeds 15 psig before drywell sprays can be initiated. Why must torus pressure exceed 15 psig prior to initiating drywell sprays?

A. This will prevent chugging in the drywell vent downcomers which could result in structural damage to the downcomers.
B. This will ensure a large enough differential pressure to open the torus to drywell vacuum breakers and reduce the torus pressure.
C. This will remove radioactivity from the noncondensibles, which must be performed prior to venting the containment.
D. The higher torus pressure will reduce the upward water force on the drywell vent header as the water in the downcomer is blown out of the downcomer.

Proposed Answer: A. This will prevent chugging in the drywell vent downcomers which could result in structural damage to the downcomers.

Explanation (Optional):
B. The torus to drywell vacuum breakers only requires a 0.5 psid to open. They do not need the high torus pressure 15 psig.
C. It is recommended that the venting is performed through the torus to scrub. It is not a requirement that this be performed prior to venting the containment.
D. The water level in the downcomer effects the upward forces on the drywell vent header. This is only a concern during the initial break because after the torus reaches 15 psig the RPV is in a lower energy state and will not be able to produce the forces that were produce prior to the primary system break.

Technical Reference(s): MIT-301.11E, pp 18

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.4.18 Knowledge of the specific basis for EOPs

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

Question History: Last NRC Exam ___________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:
Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:	55.41	__10_
	55.43	_____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	_____	__3__
	Group #	_____	_____
	K/A #	2.4.24	
	Importance Rating	_____	_3.7_

Proposed Question: S98

The loss of an operating service water pump will result in the standby pump starting on _(1)_ . If the loss of the pump results in a loss of the service water system then _(2)_ .

A. (1) trip of the running pump
(2) place the control switches for the turbine building closed loop cooling (TBCLC) pumps in pull to lock and verify that the emergency service water automatically inject into the TBCLC header.

B. (1) trip of the running pump
(2) place the control switches for the TBCLC pumps in pull to lock and verify that the emergency service water automatically inject into the TBCLC header.

C. (1) trip of the running pump or low service water header pressure of 75 psig
(2) manually scram the reactor and place the reactor building closed loop cooling (RBCLC) pumps in pull to lock.

D. (1) trip of the running pump or low service water header pressure of 85 psig
(2) manually scram the reactor and place the RBCLC pumps in pull to lock.

Proposed Answer: C. (1) trip of the running pump or low service water header pressure 75 psig
(2) manually scram the reactor and place the RBCLC pumps in pull to lock.

Explanation (Optional): The standby pump starts at a service water discharge header pressure of 75 psig or trip of the running pump. IF a complete loss of service water occurs then the reactor is scrammed and the RBCLC pumps are placed in PTL to allow ESW to cool the area temperatures in the crescent area.

Technical Reference(s): AOP-10, loss of service water and OP-42 Service Water

Proposed references to be provided to applicants during examination: None

Learning Objective: Knowledge of loss of cooling water procedures. RO 3.3 / SRO 3.7

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

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	_10__
	55.43	__5__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	__3__	__3__
Group #	_____	_____
K/A #	2.4.32	
Importance Rating	_3.3_	_3.5_

Proposed Question: 99 / 100

The reactor is at 75% power on a high rod line and is being rased to 100% power with recirculation flow. Torus cooling is inservice and personnel are being staged for the performance of ST-4N, "HPCI Quick-Start, Inservice and Transient Monitoring Test (IST)." Once the personnel are staged power ascension will be stopped for HPCI testing. A moment ago, the operators reported that the following annunciators went dark without re-flash.

(1) 09-4-1-26 CORE SPRAY OR RHR PUMP RUNNING
(2) 09-5-2-2 ROD WITHDRAWAL BLOCK

The APRM hi lights are still flashing and the RHR system is still operating in torus cooling. EPIC computer alarm LOSS OF PWR INTERPOSING RLY SYS has occurred. What has occurred to the annunciators on the 09-3 through the 09-8 panels and what actions must be taken?

A. The AC annunciator power has been lost and automatically transferred to DC power, the annunciators must be reset from panels IR-1 & IR-2 in the relay room. Power ascension and testing may continue while the annunciators are reset.
B. The AC annunciator power has been lost and must be manually transferred to DC power at panels IR-1 & IR-2 in the relay room. The power ascension and all testing will be stopped during the manual transfer to DC power.
C. The AC and DC annunciator power has been lost. Power ascension and all testing will be stopped.
D. The AC and DC annunciator power has been lost. Power ascension will be stopped; however, HPCI testing should be performed while maintenance restores the annunciators.

Proposed Answer: C. The AC and DC annunciator power has been lost. Power ascension and all testing will be stopped.

Explanation (Optional):
A. If AC power is lost and automatically transfers to DC power the annunciators do not have to be reset.
B. If AC power fails the transfer to DC backup power is automatic.
D. If annunciators are lost all power ascension and test must be stopped in accordance with AOP-65.

Technical Reference(s):AOP-65

Proposed references to be provided to applicants during examination: None

Learning Objective: 2.4.32 Knowledge of operator response to loss of all annunciators.

Question Source:
Bank # ________
Modified Bank # _______ (Note changes or attach parent)
New ___X___

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 __10_

55.43 __5__

Comments:

Examination Outline Cross-reference: Level

	RO	SRO
Tier #	_____	__3__
Group #	_____	_____
K/A #	2.4.48	
Importance Rating	_____	_3.8_

Proposed Question: S100

While operating at 90% reactor power, an event causes the following conditions to exist

- OFF GAS LINE PRESS HI annunciator 09-6-1-7 alarming.
- OFF GAS LINE TEMP HI annunciator 09-6-1-15 alarming.
- Steam Jet Air Ejector Supply Valve (29PCV-107) close.
- Condenser Isolation Valves (38AOV-113A and B) close.

Which one of the following automatic actions could also be expected to occur and what actions must be taken?

A. MSIV closure due to low Main Steam Line pressure and execute EOP-2, "RPV Control."
B. Main and Reactor Feed Pump turbine trips on high RPV level and execute AOP-1, "Scram."
C. RWR pumps trip and ARI initiation due to high RPV pressure and execute EOP-3, "failure to scram."
D. Turbine Building Ventilation isolation due to high building airborne activity and execute AOP-1, "Scram."

Proposed Answer: D. Turbine Building Ventilation isolation due to high building airborne activity and execute AOP-1, "Scram."

Explanation (Optional):

Technical Reference(s):AOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. RO 3.5 / SRO 3.8

Question Source: Bank # FitzPatrick Requalification Question Number 27101004B01C Rev. 2

While operating at 90% reactor power, an event causes the following conditions to exist

- OFF GAS LINE PRESS HI annunciator 09-6-1-7 activates.
- OFF GAS LINE TEMP HI annunciator 09-6-1-15 activates.
- Steam Jet Air Ejector Supply Valve (29PCV-107) close.
- Condenser Isolation Valves (38AOV-113A and B) close.

Which one of the following automatic actions could also be expected to occur

A. MSIV closure due to low Main Steam Line pressure.
B. Main and Reactor Feed Pump turbine trips on high RPV level.
C. RWR pumps trip and ARI initiation due to high RPV pressure.
D. Turbine Building Ventilation isolation due to high building airborne activity.

Question History: Last NRC Exam ____________
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 _____
55.43 __5__

Comments:

ATTACHMENT 5
REACTOR COOLANT CHEMISTRY REQUIREMENTS

1.1 Limiting Conditions for Operation

1.1.A Coolant Chemistry

Applicability: Applies to the operating status of the Reactor Coolant System.

Objective: To assure integrity and safe operation of the Reactor Coolant System.

Specification:

1. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 lb/hr except as specified in 1.1.A.2:
   Conductivity 2 μmho/cm
   Chloride ion 0.1 ppm

2. For reactor startups the maximum value for conductivity shall not exceed 10 μmho/cm and the maximum value for chloride ion concentration shall not exceed 0.1ppm, for the first 24 hours after placing the reactor in the power operating condition. During reactor shutdowns, specification 1.1.A.3 will apply.

3. Except as specified in 1.1.A.2 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 lb/hr and during reactor shutdowns.
   Conductivity 5 μmho/cm
   Chloride ion 0.5 ppm

4. If Specification 1.1.A cannot be met, the reactor shall be placed in a cold condition within 24 hours.

2.1 Surveillance Requirements

2.1.A Coolant Chemistry

Applicability: Applies to the periodic examination and testing requirements for Reactor Coolant Chemistry.

Objective: To determine the condition of the Reactor Coolant Chemistry.

Specification:

1. During startups and at steaming rates below 100,000 lb/hr, and when the conductivity of the reactor coolant exceeds 2 μmhos/cm, a sample of reactor coolant shall be taken every 4 hr and analyzed for conductivity and chloride content.

2. a. With steaming rates greater than or equal to 100,000 lb/hr, a reactor coolant sample shall be taken at least every 96 hours and whenever the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

   b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.

ATTACHMENT 5
REACTOR COOLANT CHEMISTRY REQUIREMENTS

1.1 and 2.1 BASES (cont'd)

Coolant Chemistry

Materials in the Reactor Coolant System are primarily 304 stainless steel and Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of the stainless steel. The attached graph, Fig. 1, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup, and hot standby. During these periods with steaming rates less than 100,000 lb/hr, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Fig. 1 are not exceeded. At steaming rates of at least 100,000 lb/hr, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal ranges. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This is not necessarily the case. Conductivity could be high due to the presence of a neutral salt; e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are, in fact, high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the condition include operation of the Reactor Cleanup System, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the Reactor Water Cleanup System to reestablish the purity of the reactor coolant. Noble Metal Chemical Addition (NMCA) is a process of applying noble metals currently platinum (Pt) and rhodium (Rh), to the wetted surfaces of the reactor vessel and recirculation system to reduce the feedwater hydrogen concentration required for IGSCC mitigation of reactor internal components. After an initial application, NMCA will be reapplied in future years as deemed necessary to maintain a catalytic surface on the wetted components.

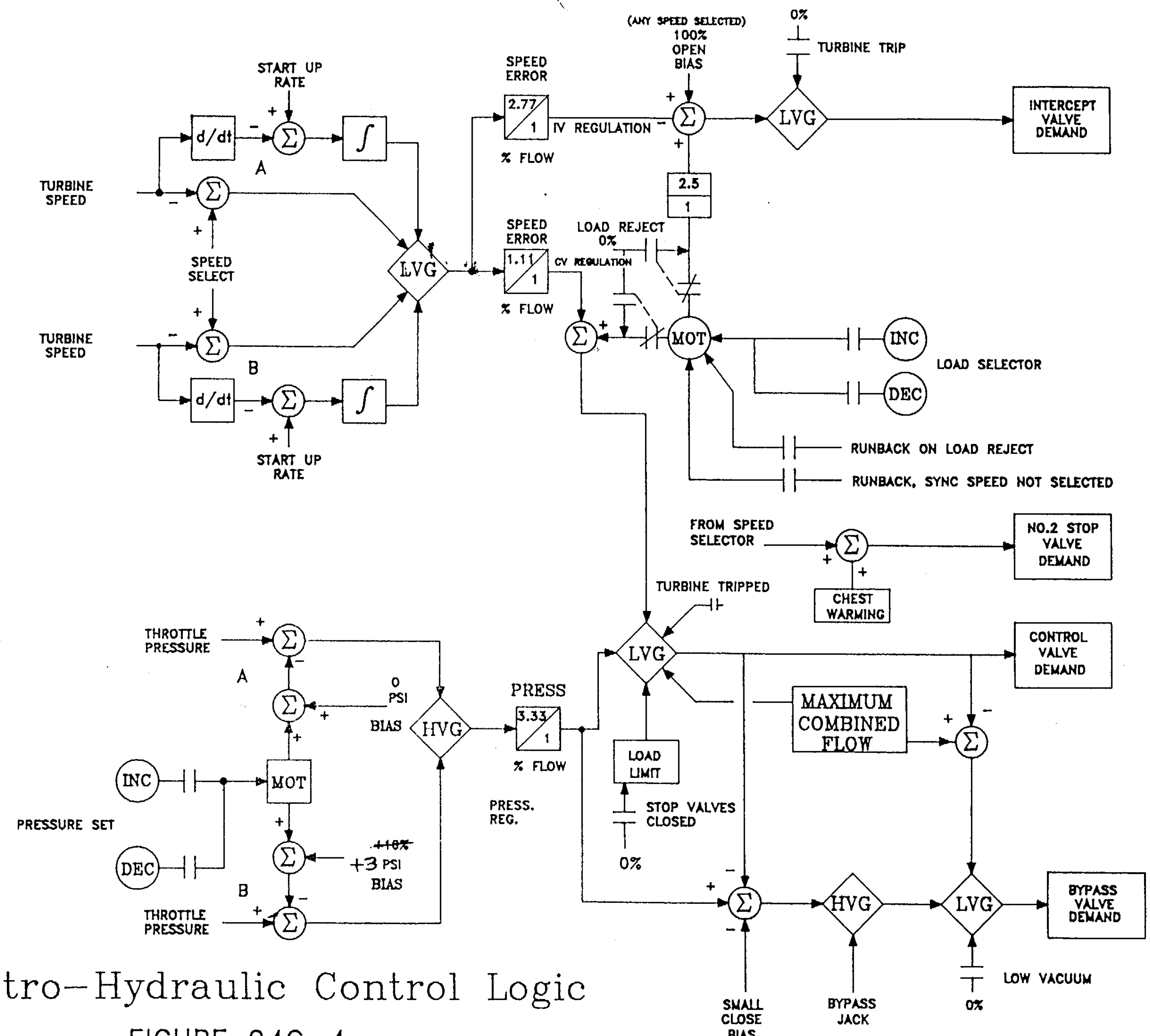
ATTACHMENT 5
REACTOR COOLANT CHEMISTRY REQUIREMENTS

1.1 and 2.1 BASES (cont'd)

Coolant Chemistry

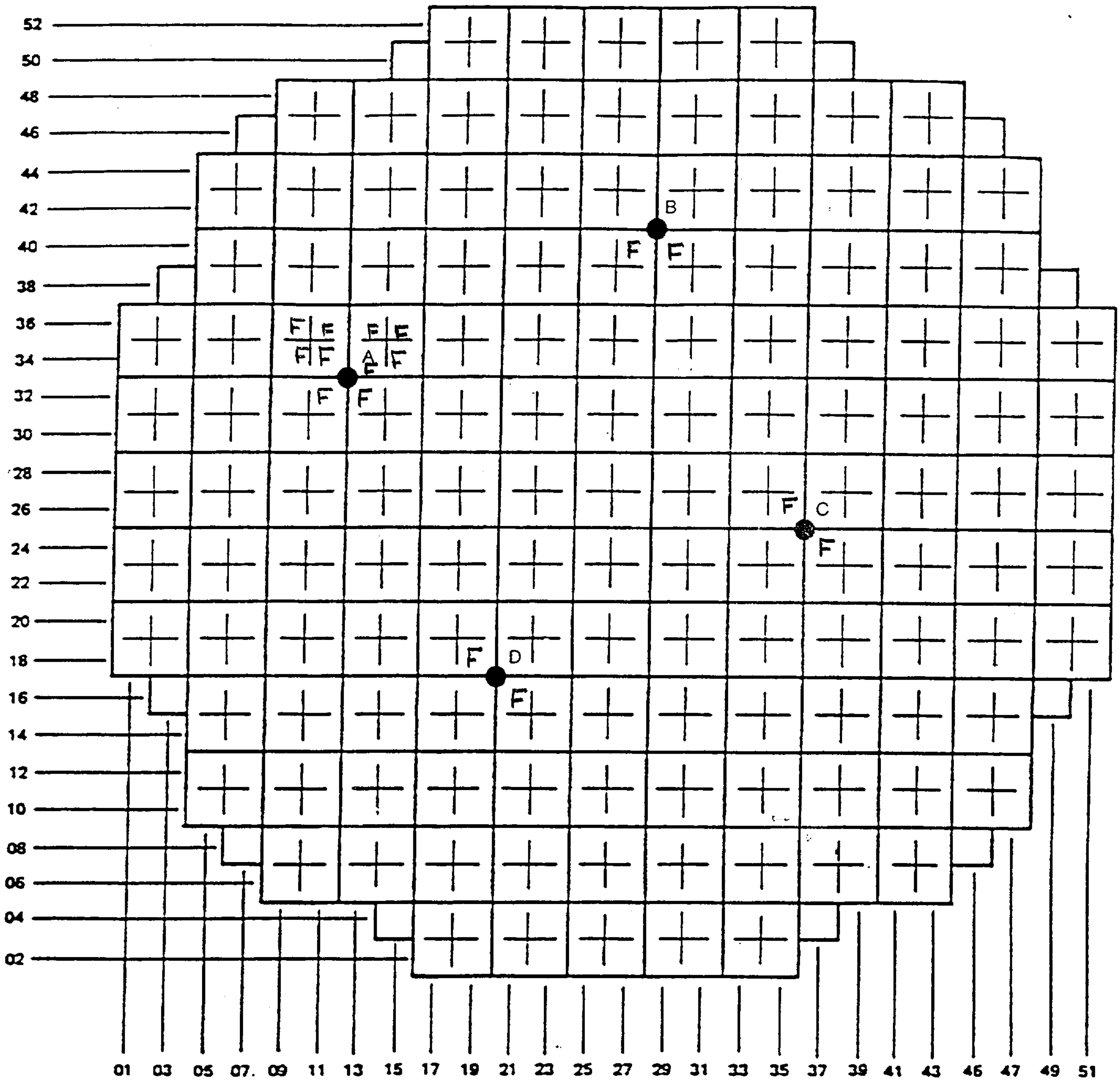
During startup periods, which are in the category of less than 100,000 lb/hr, conductivity may exceed 2 μmho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 μmho/cm (other than short-term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm. During infrequent periods of the Noble Metal application process, it is acceptable for conductivity to increase up to 20 micro-mho/cm. This relaxation is acceptable because the reduction in crack growth over subsequent operating periods outweighs the increase in crack growth over the short duration of the noble metals application (<72 hours). The anticipated conductivity increase is due to factors unrelated to the increase in trace amounts of sulfates and chlorides contained in the noble metal compounds.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses of the reactor coolant required by Technical Specification 4.6.C.1 may be performed by a gamma scan.



Electro-Hydraulic Control Logic

FIGURE 94C-4



● SRM LOCATIONS

F Fuel Assembly

DETECTOR ASSEMBLY IN-CORE LOCATIONS