Examination Outline Cross-Reference	Level	RO
203000 RHR/LPCI: Injection Mode (Plant Specific)	**Tier**	2
	Group	1
	K/A #	203000.A2.03
A.2 Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	**Rating**	3.2
A2.03 Valve closures	Rev / Date	2

Proposed Question: RO-01

RPV level is -150 inches and steady. RHR-P-2C is currently injecting full flow and is the **ONLY** available injection source.

The control switch for RHR-V-4C (RHR-P-2C's suction valve) is inadvertently taken to close **AND** fully closes.

Which of the following is correct?

A. RHR-P-2C trips. Re-open RHR-V-4C, restart RHR-P-2C and recommence RPV injection.

B. RHR-P-2C does **NOT** trip. Reopen RHR-V-4C and recommence RPV injection.

C. RHR-P-2C trips. Disable the restart of RHR-P-2C by pulling the control power fuses.

D. RHR-P-2C does **NOT** trip. Stop RHR-P-2C and disable the restart of RHR-P-2C by pulling the control power fuses.

Proposed Answer: B

Explanation (Optional):

A (incorrect) Indicates RHR-P-2C trips (which it does not). The remainder of the distractor is correct.

B (CORRECT) RHR-P-2C does not have a trip associated with the position of the RHR-V-4C (Pumps suction valve). The ARP directs reopening RHR-V-4C and if it cannot be opened, then disabling the starting of RHR-P-2C. RHR-P-2C should be restarted as it is the only injection source.

C (incorrect) Indicates RHR-P-2C trips. The pump should also be restarted. Only if RHR-V-

4C cannot be reopened should disabling the pump start be performed. Control power fuses would be directed pulled by the expected low discharge pressure alarm ARP.

D (incorrect) RHR-P-2C does not trip – this is correct. RHR-P-2C should not be disabled from starting as it is the only RPV injection source. Control power fuses would be directed pulled by the expected low discharge pressure alarm ARP.

Technical Reference(s): SD000198 Rev. 015 Page 11
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11586 (As available)

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 H / 2

10 CFR Part 55 Content: 55.41 5
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) **A.1 Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including:** A.1.06 Reactor temperatures (moderator, vessel, flange)	**Tier**	2
	Group	1
	K/A #	205000.A.1.06
	Rating	3.7
	Rev / Date	2

Proposed Question: RO-02

During operation of RHR B in Shutdown Cooling, RHR-V-3B (RHR HX Shell Side Outlet Valve) **AND** RHR-V-48B (RHR HX Shell Side Bypass Valve) are both 50% OPEN. RHR-V-42B (LPCI Isolation valve) is danger tagged closed. RPV temperature has been at 110°F for the past 20 minutes.

If a High Drywell Pressure GT 1.68 psig signal initiates, which of the following is correct?

A. RPV temperature will raise due to RHR-V-48B opening.

B. RPV temperature will lower due to RHR-V-48B closing.

C. RPV temperature will lower due to RHR-V-3B opening.

D. RPV temperature will raise due to RHR-V-3B closing.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) On a LPCI initiation signal, RHR-V-48B opens and RHR-V-3B does not move during on a high DW pressure signal. Other valves in the RHR system do get close signals on high Drywell pressure signals. RHR-V-48B opening causes less flow to pass through the RHR heat exchanger. The less flow through the heat exchanger, RPV temperature increases.

B (incorrect) Choice indicates RPV temperature decreases and RHR-V-3B opens.

C (incorrect) Choice indicates RPV temp increases but RHR-V-3B closes.

D (incorrect) Choice indicates RPV temp increases but RHR-V-3B closes.

Technical Reference(s): SD000198 Rev. 015 Page 12, 20, 25 and 26
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7728 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 5
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
209001 Low Pressure Core Spray System	**Tier**	2
	Group	1
K4. Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following:	**K/A #**	209001.K4.06
	Rating	2.6
K4.06 Adequate pump net positive suction head	Rev / Date	2

Proposed Question: RO-03

Which of the following design features help to maintain adequate Net Positive Suction Head (NPSH) for the LPCS pump?

1. LPCS suction strainer size
2. LPCS pump elevation in reference to the Suppression Pool
3. LPCS water leg pump keeping the suction piping filled with water

A. 1 only

B. 2 only

C. 1 and 2

D. 1, 2, and 3

Proposed Answer: C

Explanation (Optional):
Choice 1 is correct as per the CGS FSAR the LPCS pump is located in the reactor building sufficiently below the water level in the suppression pool to ensure a flooded pump suction.

Choice 2 is correct as per the CGS FSAR the LPCS pump has sufficient NPSH with the containment at atmospheric pressure and post-accident debris entrained on the beds of the suction strainers.

Choice 3 is incorrect but plausible as the LPCS water leg pump maintains the suction and discharge piping filled which aids in adequate NPSH but is designed to prevent damage from fluid hammer. RCIC system keep fill is designed to maintain NPSH and it discharges to the suction of the pump making it plausible that the same design could hold true for the LPCS system.

A (incorrect) Does not include choice 2 which is also correct - as per the CGS FSAR the LPCS pump has sufficient NPSH with the containment at atmospheric pressure

and post-accident debris entrained on the beds of the suction strainers.

B (incorrect) Does not include choice 1 which is correct also - as per the CGS FSAR the LPCS pump is located in the reactor building sufficiently below the water level in the suppression pool to ensure a flooded pump suction.

C (CORRECT) Includes both choices 1 and 2 - Choice 1 is correct as per the CGS FSAR the LPCS pump is located in the reactor building sufficiently below the water level in the suppression pool to ensure a flooded pump suction.
Choice 2 is correct as per the CGS FSAR the LPCS pump has sufficient NPSH with the containment at atmospheric pressure and post-accident debris entrained on the beds of the suction strainers.

D (incorrect) Includes choice 3 which is not correct but is plausible - Choice 3 is incorrect but plausible as the LPCS water leg pump maintains the suction and discharge piping filled with aids in adequate NPSH but is designed to prevent damage from fluid hammer. This answer was selected by licensed operators during validation enhancing the plausibility of the distractor.

Technical Reference(s): FSAR 6.3.2.2.3
(Attach if not previously provided, including version/revision number) _____

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11586 (As available)

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 F / 2
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
209002 High Pressure Core Spray System (HPCS)	**Tier**	2
	Group	1
	K/A #	209002.K3.03
K.3 Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) will have on following:	**Rating**	3.9
	Rev / Date	2
K3.03 Adequate core cooling: BWR-5,6		

Proposed Question: RO-04

HPCS and RCIC are operating following a reactor scram both taking a suction from the Suppression Pool with the following conditions:

RCIC flowrate is 650 GPM
HPCS flowrate is 7000 GPM
RPV pressure is 850 psig steady
RPV level -205 inches and up slow

Debris partially plugs the HPCS suction strainer resulting in the following conditions:

RCIC flowrate is 650 GPM
HPCS flowrate is 5500 GPM
RPV pressure is 850 psig steady
RPV level -203 inches and steady

Can adequate core cooling be assured under these conditions **AND** why or why not?

A. Yes. RPV level is being maintained above level required for spray cooling with HPCS and RCIC providing sufficient flow to ensure adequate core cooling.

B. Yes. RPV level is above level required for spray cooling and steam cooling with injection will ensure adequate core cooling.

C. No. HPCS is not providing an adequate amount flow for spray cooling, and RPV level is below the level for steam cooling with injection to ensure adequate core cooling.

D. No. HPCS and RCIC are providing sufficient spray flow, but RPV level is below level required to ensure adequate core cooling.

Proposed Answer: C

Explanation (Optional):

A (incorrect) HPCS or LPCS by themselves must provide greater than 6000 GPM for spray cooling to be effective.

B (incorrect) RPV level must be greater than -183 inches with injection to ensure adequate core cooling.

C (CORRECT) HPCS is not providing adequate spray flow and RPV level must be greater than -183 inches with injection to ensure adequate core cooling.

D (incorrect) HPCS or LPCS by themselves must provide greater than 6000 GPM for spray cooling to be effective.

Technical Reference(s): PPM 5.0.10 Rev. 019 Pages 117 and 118
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8018 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
209002 High Pressure Core Spray System (HPCS)	**Tier**	2
	Group	1
	K/A #	209002.K6.01
Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS):	**Rating**	3.6
	Rev / Date	2
K6.01 Electrical power.		

Proposed Question: RO-05

Which of the following describes the effect on the High Pressure Core Spray system components if MC-4A loses power just prior to a HPCS low RPV level initiation signal occurring?

HPCS-P-1…..

A. starts. HPCS-V-4 (Injection valve) and HPCS-V-12 (Min flow valve) remain closed due to a loss of power.

B. starts. HPCS-V-4 (Injection valve) opens to inject to the vessel. HPCS-V-12 (Min flow valve) opens and closes based on injection flow.

C. does **NOT** start. HPCS-V-4 (Injection valve) and HPCS-V-12 (Min flow valve) remain closed due to the loss of power.

D. does **NOT** start. HPCS-V-4 (Injection valve) opens to inject to the vessel. HPCS-V-12 (Min flow valve) opens due to no injection flow.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) The power supply for the initiation logic is S1-HPCS and the power supply for HPCS-P-1 is SM-4 – therefore HPCS-P-1 will start. MC-4A powers all HPCS system valves and HPCS-P-3.

B (incorrect) HPCS-V-4 and HPCS-V-12 loses power and will not open.

C (incorrect) HPCS-P-1 does start.

D (incorrect) HPCS-P-1 does start, HPCS-V-4 and HPCS-V-12 loses power and will not open.

Technical Reference(s): SD000174 Rev. 013 Page 21
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5431 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
211000 Standby Liquid Control System	Tier	2
	Group	1
K2 Knowledge of electrical power supplies to the following:	K/A #	211000.K2.02
	Rating	3.1
K2.02 Explosive valves	Rev / Date	2

Proposed Question: RO-06

A loss of power to which of the following would prevent SLC-V-4A (Squib Injection valve) from opening?

A. MC-7A

B. MC-8A

C. MC-7B

D. MC-8B

Proposed Answer: C

Explanation (Optional):

A (incorrect) MC-7A is in the same division (Div. 1) and is a critical bus, but does not provide power to the SLC valves. Plausible because it is an "A" designator.

B (incorrect) MC-8A is the opposite division (Div. 2) and is a critical bus, but does not provide power to the SLC valves. Plausible because it is an "A" designator.

C (CORRECT) SLC-V-4A is powered from MC-7B.

D (incorrect) MC-8B is the power supply to SLC-V-4B.

Technical Reference(s): SD000172 Rev. 013 Page 24
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 ___F / 2___

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7____

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
212000 Reactor Protection System	**Tier**	2
	Group	1
K1 Knowledge of the physical connections and/or cause effect relationships between REACTOR PROTECTION SYSTEM and the following:	**K/A #**	212000.K1.06
	Rating	3.5
K1.06 Control rod drive hydraulic system.	Rev / Date	2

Proposed Question: RO-07

Which of the following explains how a trip of the Reactor Protection System affects the following components?

	Scram Air Pilot Valves	Backup Scram Valves
A.	Energize	De-Energize
B.	Energize	Energize
C.	De-energize	De-energize
D.	De-energize	Energize

Proposed Answer: D

Explanation (Optional):

A (incorrect) See D

B (incorrect) See D

C (incorrect) See D

D (CORRECT) RPS de-energizes the scram air pilot valves and energizes the backup scram valves.

Technical Reference(s): SD000161 Rev. 017 Page 4
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7682 (As available)

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 F / 2

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 6

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
215003 Intermediate Range Monitor (IRM) System	**Tier**	2
	Group	1
	K/A #	215003.G2.1.28
G2 Intermediate Range Monitor System:	**Rating**	4.1
G2.1.28 Knowledge of the purpose and function of major system components and controls.	Rev / Date	2

Proposed Question: RO-08

Control rods are being withdrawn for a plant startup in Mode 2 with the following conditions:

- All IRMs are fully inserted
- The SRM/IRM Detector Positioner POWER ON pushbutton is illuminated
- The IRM-C SELECTED pushbutton is illuminated

What is the status of the RETRACT PERMIT light for IRM-C, **AND** what will occur if the DRIVE OUT pushbutton is depressed?

The RETRACT PERMIT light for IRM-C is….

A. extinguished.
The detector will fully withdraw and a rod block will be generated.

B. extinguished.
The detector will **NOT** withdraw and **NO** rod block will be generated.

C. illuminated
The detector will fully withdraw and **NO** rod block will be generated.

D. illuminated
The detector will fully withdraw and a rod block is generated.

Proposed Answer: A

Explanation (Optional):

This question is testing the function of the detector retract pushbutton and the purpose of the retract permit light.

A (CORRECT) With the plant in Mode 2 as stated in the stem, the RETRACT PERMIT light will be extinguished. This light does not refer to the ability to retract the IRM as its name suggests, but to the ability to withdraw the IRM without receiving a rod block. Therefore, with the light extinguished, the IRM may be withdrawn, but

will cause a rod block.

B (incorrect) The detector will withdraw. This distractor is plausible if the candidate believes the function of the retract permit circuit is to indicate when the conditions to withdraw the detector are met.

C (incorrect) The light will be extinguished. This distractor contains the indications and conditions that would result in Mode 1.

D (incorrect) This distractor is plausible, because it describes the function of the SRM retract permit circuit.

Technical Reference(s): SD000138 Rev. 010 Page 12 and 13; Simulator
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5456 (As available)

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 H / 4

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
215004 Source Range Monitor (SRM) System	**Tier**	2
	Group	1
	K/A #	215004.A4.01
A4 Ability to manually operate and/or monitor in the control room:	**Rating**	3.9
A4.01 SRM count rate and period	Rev / Date	2

Proposed Question: RO-09

During a reactor startup, the SRM upscale **AND** SRM period fast alarms were received **AND** cleared immediately. All SRM's are reading between 3000 cps and 4000 cps.

What is the status of the upscale alarm lights for the SRM channel(s) on H13-P603, and H13-P606/P633 for the channel(s) that caused the alarm?

A. P603 – Clear
P606/P633 – Clear

B. P603 – Clear
P606/P633 – Lit

C. P603 – Lit
P606/P633 – Clear

D. P603 – Lit
P606/P633 – Lit

Proposed Answer: B

Explanation (Optional):

H13-P603 –There is an upscale alarm lamp for each SRM on P603; these alarm lights automatically extinguish when the associated SRM reading is below 1 x 10^5 cps. The stem gives SRM readings that are all below the level that would cause one or more of these alarm lights to remain lit. The common annunciator (SRM upscale) alarms and must be manually acknowledged and reset from P603, so it is plausible that the individual SRM lights could still be lit. Additionally, the back panel alarm lights on the SRM drawers do seal in.

Seal in Trip lamps for each SRM's are located on H13-P606 (SRM channels A and B) and H13-P633 (SRM channels C and D). These lights remain locked in and must be manually reset with a switch on the individual SRM instrument drawers.

A (incorrect) P603 - Correct.
P606/P633 – Incorrect

B (CORRECT) P603 - Correct.
P606/P633 – Correct.

C (incorrect) P603 - Incorrect.
P606/P633 – Incorrect.

D (incorrect) P603 - Incorrect.
P606/P633 – Correct.

Technical Reference(s): SD000132 Rev. 012 page 16
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11997 (As available)

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 F / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
215005 Average Power Range Monitor/Local Power Range Monitor System	**Tier**	2
	Group	1
	K/A #	215005.A3.06
A3 Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including:	**Rating**	3.0
A3.06 Maximum disagreement between flow comparator channels	Rev / Date	2

Proposed Question: RO-10

Of the following values, which is the largest disagreement between consecutive APRM flow comparator channels that will **NOT** generate a rod withdrawal block signal?

A. 4%

B. 9%

C. 14%

D. 19%

Proposed Answer: B

Explanation (Optional):

A (incorrect) At 5%, an APRM downscale rod block , but with the mode switch in run this would not be received in RUN. This is not the flow unit comparator rod block setpoint.

B (CORRECT) A 10% difference will produce a rod block signal from the flow comparator.

C (incorrect) 15% is an APRM power trip setpoint if the mode switch is not in run. This is not the flow unit comparator rod block setpoint.

D (incorrect) Answer is a factor of 5 higher than other distractors to maintain consistency.

Technical Reference(s): SD000149 Rev. 014 page 9
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11654 (As available)

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 F / 3

10 CFR Part 55 Content: 55.41 7

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
217000 Reactor Core Isolation Cooling System (RCIC) **K1 Knowledge of the physical connections and/or cause – effect relationships between REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and the following:** K1.01 Condensate storage and transfer system	**Tier**	2
	Group	1
	K/A #	217000.K1.01
	Rating	3.5
	Rev / Date	2

Proposed Question: RO-11

The Reactor Core Isolation Cooling (RCIC) system has been manually started for vessel pressure control **AND** is operating CST (Condensate Storage Tank) to CST.

Select the statement below that identifies the RCIC system response if CST level decreases to 1' 6".

A. The CST suction valve will close causing a RCIC turbine trip on low pump suction pressure.

B. RCIC will be operating with a Suppression Pool to Suppression Pool flow path through the RCIC full flow test line.

C. The RCIC pump suction valves swap resulting in RCIC taking a suction from the Suppression Pool and transferring water to the CSTs.

D. RCIC will take a suction from the Suppression Pool and discharge to the Suppression Pool through the RCIC minimum flow valve.

Proposed Answer: D

Explanation (Optional):

A (incorrect) A suction flow path is maintained throughout the valve swap as one does not start to close until the other is full open.

B (incorrect) The full flow test line discharges to the CSTs.

C (incorrect) The valves in the full flow test line close when the Suppression Pool suction valve is open.

D (CORRECT) At a CST level of 1'10" the RCIC pump suction valves swap. When the Suppression Pool suction valve is full open the valves in the full flow test line to the CST close.

Technical Reference(s): SD000180 Rev. 016 pages 18, 19, 20, 21 and 22
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5724 (As available)

Question Source: Bank # LO00803
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam Not used on 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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
218000 Automatic Depressurization System	Tier	2
	Group	1
K2 Knowledge of electrical power supplies to the following:	K/A #	218000 K2.01
	Rating	3.1
K2.01 ADS logic	Rev / Date	2

Proposed Question: RO-12

An event occurs that results in a loss of DC bus S1-2.

Which of the following describes the effect this loss has on the ADS SRVs?

A. Only the 'A' solenoids lose power.

B. Only the 'B' solenoids lose power.

C. The 'A' and 'B' solenoids lose power.

D. The 'A' and 'C' solenoids lose power.

Proposed Answer: B

Explanation (Optional):

A (incorrect) The 'A' solenoid is powered from DP-S1-1A and remain powered.

B (CORRECT) A loss of bus S1-2 causes DP-S1-2A to lose power causing a loss of power to the 'B' solenoids.

C (incorrect) The 'A' solenoid is powered from DP-S1-1A and remain powered.

D (incorrect) The 'A' and the 'C' solenoids are powered from DP-S1-1A and remain powered.

Technical Reference(s): SD000128 Rev. 012 Pages 29 and 30;
(Attach if not previously provided, including version/revision number) SD000188 Rev. 010 Figure 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5077 (As available)

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 ___F / 2___

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _7_____

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off	**Tier**	2
	Group	1
	K/A #	223002 K3.11
K3 Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following:	**Rating**	2.8
	Rev / Date	2
K3.11 Plant ventilation		

Proposed Question: RO-13

With Columbia operating in MODE 1, ROA-FN-1B (Reactor Building HVAC Supply Fan), **AND** REA-FN-1B (Reactor Building HVAC Exhaust Fan), are running **AND** REA-RIS-609C (Reactor Building Exhaust Plenum Radiation Monitor) is INOP (switch is out of operate).

Which of the following explains the effect if REA-RIS-609D receives a Hi-Hi trip signal?

A. ROA-FN-1B and REA-FN-1B trip. **ONLY** the 'A' Standby Gas Treatment train initiates.

B. ROA-FN-1B and REA-FN-1B continue to operate. **ONLY** the 'B' Standby Gas Treatment train initiates.

C. ROA-FN-1B and REA-FN-1B trip. **BOTH** the 'A' and 'B' Standby Gas Treatment trains initiate.

D. ROA-FN-1B and REA-FN-1B continue to operate. **NEITHER** Standby Gas Treatment train initiates.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Both trains of SGT will start. (See C for Explanation.)

B (incorrect) ROA-FN-1B and REA-FN-1B trip. (See C for Explanation.)

C (CORRECT) A trip of Group 3 logic for RB Exhaust Plenum radiation monitors is arranged such that two monitors must trip to actuate the inboard or outboard isolation (A&B OR C&D). INOP, Downscale, or Upscale will all cause a Z isolation signal to be sent. This condition for C & D initiates closure of the corresponding inboard valves and initiates startup of SGT Train B Lead Fan and Train A Lag

Fan. Both trains of SGT will start.

D (incorrect) Both trains of SGT will start and ROA-FN-1B and REA-FN-1B trip. (See C for Explanation.)

Technical Reference(s):
(Attach if not previously provided, including version/revision number)

SD000173 Rev. 014 page 17, 23 and 24,
SD000147 Rev. 013Page 20 & 21,
EWD-108E-003, EWD-108E-005, EWD-108E-007

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5598 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
239002 Relief/Safety Valves	**Tier**	2
	Group	1
K4 Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following:	**K/A #**	239002.K4.03
	Rating	3.1
K4.03 Prevents siphoning of water into SRV discharge piping and limits loads on subsequent actuation of SRV's	Rev / Date	2

Proposed Question: RO-14

During potential subsequent actuations of an SRV, which of the following correctly describes the design feature that limits BOTH piping stress **AND** containment loading?

A. The X shaped quencher that is installed at the end of each SRV tailpipe.

B. The Suppression Pool minimum water level of 19'2".

C. The vacuum breakers that are installed in each downcomer, located below the Drywell floor.

D. The vacuum breakers that are installed in each SRV tailpipe, located above the Drywell floor.

Proposed Answer: D

Explanation (Optional):

A (incorrect) The quencher will prevent unstable steam condensation oscillations which could damage the containment vessel. This does not prevent piping stress on the SRV Tailpipe.

B (incorrect) Suppression pool level will condense the steam limiting pressure buildup in the Wetwell. This will not prevent a vacuum from forming in the SRV tailpipe after a SRV lift.

C (incorrect) These are the Wetwell to Drywell vacuum breakers which limit the upward force on the Drywell floor following a LOCA when the Drywell pressure drops below the Wetwell pressure. This does not prevent piping stress on the SRV Tailpipe.

D (CORRECT) There are vacuum breakers installed above the drywell floor that are designed to limit containment loading due to multiple actuations of an SRV.

Technical Reference(s): SD000128 Rev. 012 page 14
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11690, 11745 (As available)

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 F / 2
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
259002 Reactor Water Level Control System	**Tier**	2
	Group	1
K5 Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM:	**K/A #**	259002 K5.01
	Rating	3.1
K5.01 GEMAC/Foxboro/Bailey controller operation	Rev / Date	2

Proposed Question: RO-15

Following a scram, RPV water level is being maintained at +36 inches using the Startup Flow Control Valves (RFW-FCV-10A/B) in Automatic.

A sudden and immediate loss of the output signal from the Startup RPV Level Controller (RFW-LIC-620) occurs.

What will happen to RPV level as a result of this failure?

RPV Level will ….

A. remain at +36 inches based on the last known RPV level setpoint received by the PLC modulating RFW-FCV-10A/B.

B. increase because the PLC loses its level input signal and fully opens RFW-FCV-10A/B.

C. decrease because the automatic level signal setpoint will fail downscale causing RFW-FCV-10A/B to close fully.

D. decrease to +18 inches based on setpoint setdown and modulate RFW-FCV-10A/B to maintain +18 inches.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) S/U valve demand continues from the PLC based on the last known setpoint input from RFW-LIC-620.

B (incorrect) PLC receives level signal directly from the level detectors.

C (incorrect) S/U valve demand continues from the PLC based on the last known setpoint

input from RFW-LIC-620.

D (incorrect) Setpoint setdown is a function of the Master Controller and the RFP's speed.

Technical Reference(s): SD000157 Rev. 16 page 9
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11704 (As available)

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 F / 4
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
261000 Standby Gas Treatment System	**Tier**	2
	Group	1
K6 Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM :	**K/A #**	261000.K6.01
	Rating	2.9
K6.01 A.C. electrical distribution	Rev / Date	2

Proposed Question: RO-16

SGT 'A' has been running for 20 minutes due to an automatic initiation signal being present.

If a lockout on SM-7 were to occur, which of the following explains the response of the SGT Train 'A' system?

The lead fan loses power. The lag fan then receives a start signal, starts, ….

A. but trips on low flow. The lag fan repeatedly starts and trips on low flow.

B. and maintains Secondary Containment pressure.

C. but trips on low flow and does not restart.

D. and is unable to maintain Secondary Containment pressure.

Proposed Answer: B

Explanation (Optional):

A (incorrect) The low flow condition does not occur on the lag fan because its inlet valve is normally open and its discharge valve is powered from MC-8B-B. This allows the lag fan to take a suction from the common filter train and discharge to the elevated release. This answer is plausible since a component in the lag train is powered from the deenergized bus (the normally open inlet valve) and loss of a different power supply (PP-8AA) would result in the repetitive start and trip sequence described in this distractor.

B (CORRECT) MC-7B-B (powered from SM-7) powers the lead fan (SGT-FN-1A1) and the lead fan's valves. When power is lost the lead fan loses power and stops. The lead fan's discharge valve also loses power and remains in the open position. The lag fan then gets a start signal on low flow and starts and automatically maintains secondary containment pressure using the lag train's DPIC.

C (incorrect) See A. This distractor is plausible for the reasons given for distractor A. The lack of a restart following the proposed low flow fan trip is plausible since the SGT *lead* fans do not restart following low flow trips.

D (incorrect) See B. The inability of the lag fan to maintain secondary containment pressure is plausible since a potential reverse flow path through the idle lead fan exists, but the lead fan's suction valve is powered from MC-8B-B and receives a close signal following the lead fan low flow trip to prevent reverse flow.

Technical Reference(s): SD000144 Rev. .015 Pages 9, 13 and 14
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5825 (As available)

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 H / 4

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
261000 Standby Gas Treatment System	**Tier**	2
	Group	1
G2 Standby Gas Treatment System:	**K/A #**	261000.G2.1.30
	Rating	4.4
G2.1.30 Ability to locate and operate components, including local controls	Rev / Date	2

Proposed Question: RO-17

A charcoal filter temperature high alarm has annunciated in the control room. OPS 2 was dispatched to investigate and has reported evidence of a fire in the SGT charcoal. The CRS directs actions per ABN-SGT-TEMP/RAD.

Which of the following is correct concerning the above?

To suppress the fire, direction is given to….

A. place the Emergency Deluge Spray valves control switch that are located on the local control panel in FLOOD.

B. place the Emergency Deluge Spray valves control switch that are located on the Control Room backpanel in FLOOD.

C. place **AND HOLD** the Emergency Deluge Spray valves control switch that are located on the local control panel in FLOOD.

D. place **AND HOLD** the Emergency Deluge Spray valves control switch that are located on the Control Room backpanel in FLOOD.

Proposed Answer: D

Explanation (Optional):

A (incorrect) Switches are in the Control Room.

B (incorrect) Switches have to be held in the flood position or the switch returns to close and the valves close.

C (incorrect) Switches are in the Control Room.

D (CORRECT) The Emergency Deluge Spray valves are controlled from the control room and have to be placed and HELD in the Flood position as they spring return to close which causes the valves to close. The valves only open in the flood position and the signal does not seal in.

Technical Reference(s): ABN-SGT-TEMP/RAD Rev. 003 Page 5
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5826 (As available)

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 F / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
262001 A.C. Electrical Distribution	**Tier**	2
	Group	1
K5. Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION:	**K/A #**	262001.K5.02
	Rating	2.6
K5.02 Breaker control	Rev / Date	2

Proposed Question: RO-18

If control power is lost to CB-7/1, which of the following is correct?

Concerning operation at the breaker cubicle, without manually charging the springs CB-7/1 can be manually …..

A. closed, opened once, and then closed again.

B. only opened once.

C. only closed once.

D. opened, closed once, and then opened again.

Proposed Answer: D

Explanation (Optional):

A (incorrect) Must know that the normal position for this breaker is closed. This is a balanced distractor for the correct answer D.

B (incorrect) The breaker can also be closed manually because the charging springs were charged before power was lost.

C (incorrect) Must know that the normal position for this breaker is closed. The breaker can also be opened manually.

D (CORRECT) When CB-7/1 closes the opening springs are charged by the closing of the breaker mechanism. The closing springs are charged via a DC motor. When the control power was lost the closing and opening springs remain charged. The opening action of the breaker will discharge the opening springs. When the breaker is manually closed, the opening springs are charged by the closing action. This will allow the opening of the breaker one final time.

Technical Reference(s): SD000182 Rev. 018 Page 29, 30, 31and 32
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5066 (As available)

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 H / 2

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
262001 A.C. Electrical Distribution	**Tier**	2
	Group	1
A1 Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including:	**K/A #**	262001.A1.03
	Rating	2.9
A1.03 Bus voltage	Rev / Date	2

Proposed Question: RO-19

The station is preparing for startup following a refueling outage. The electric plant is in a normal lineup for startup with the exception of SM-8, which is on E-TR-B (Backup Transformer) following surveillance testing. Work Control has requested the start of CW-P-1C for PMT. CW-P-A and CW-P-1B are running.

Starting CW-P-1C will cause a…

A. degraded frequency condition on E-TR-B (Backup Transformer).

B. degraded frequency condition on E-TR-S (Startup Transformer).

C. degraded voltage condition on E-TR-B (Backup Transformer).

D. degraded voltage condition on E-TR-S (Startup Transformer).

Proposed Answer: D

Explanation (Optional):

A (incorrect) Must recognize in the conditions stated in the stem that the Backup Transformer is divorced from SM-3.

B (incorrect) Frequency is not the concern for this situation.

C (incorrect) Must recognize in the conditions stated in the stem that the Backup Transformer is divorced from SM-3.

D (CORRECT) Starting a third CW Pump while E-TR-S is supplying house loads will cause an under voltage or degraded voltage on E-TR-S and the electrical buses it is powering. CW-P-1C is powered from SM-3. SM-3 normally powers SM-8. When SM-8 is powered from TR-B, breaker 8-3 is open. As a result, starting the third CW Pump will not impact E-TR-B. Frequency is determined by the 230kv grid, and is relatively constant. Starting the third pump will have minimal

impact.

Technical Reference(s): SOP-CW-START Rev. 007 Page 7; LER 84-079
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5050 (As available)

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

Question History: Last NRC Exam 2013 NRC Exam #46
(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 F / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 5
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
262002 Uninterruptable Power Supply (A.C./D.C.)	**Tier**	2
	Group	1
	K/A #	262002.A2.01
A2 Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	**Rating**	2.6
A2.01 Under voltage	Rev / Date	2

Proposed Question: RO-20

With the plant operating at 100% power, the E-IN-1 inverter voltage starts to lower.
As voltage is dropping, what is the first value of voltage that a transfer would occur **AND** to what source would it transfer?

A. 89% and MC-7A.

B. 89% and MC-7F.

C. 94% and MC-7A.

D. 94% and MC-7F.

Proposed Answer: D

Explanation (Optional):

A (incorrect) The static switch will auto transfer to the alternate AC source when any phase of the inverter drops below 5% of its normal voltage. MC-7A is the Bypass source not the Alternate source which the static switch transfers to.

B (incorrect) The static switch will auto transfer to the alternate AC source when any phase of the inverter drops below 5% of its normal voltage. Second half is correct.

C (incorrect) The static switch will auto transfer to the alternate AC source when any phase of the inverter drops below 5% of its normal voltage. MC-7A is the Bypass source not the Alternate source which the static switch transfers to.

D (CORRECT) The static switch will auto transfer to the alternate AC source when any phase of

the inverter drops below 5% of its normal voltage. The Static switch will transfer to the Alternate AC source MC-7F.

Technical Reference(s): SD000194 (UPS) Rev 11 Page 8;
(Attach if not previously provided, including version/revision number) ABN-ELEC-INV Rev 12 Page 3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5896, 5891 (As available)

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

Question History: Last NRC Exam Last NRC Exam Q#47
(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 F / 3

10 CFR Part 55 Content: 55.41 5
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
263000 D.C. Electrical Distribution	**Tier**	2
	Group	1
A3 Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including:	**K/A #**	263000.A3.01
	Rating	3.2
A3.01 Meters, dials, recorders, alarms, and indicating lights.	Rev / Date	2

Proposed Question: RO-21

Columbia is operating in Mode 1 when a ground alarm is received on Battery S1-2. CRO2 reports that the S1-2 ground detection meter indicates 0KΩ (ohms) to the POS (positive) side.

Which of the following is correct for this indication?

A. The annunciator is spurious; the meter indicates no ground on S1-2.

B. The annunciator is valid; the meter indicates a severe ground on S1-2.

C. The annunciator is spurious; the Ground Test Switch has been placed in POS (positive) and released but the reset pushbutton has **NOT** been depressed.

D. The annunciator is valid; the Ground Test Switch has been placed in POS (positive) and released but the reset pushbutton has **NOT** been depressed.

Proposed Answer: B

Explanation (Optional):

A (incorrect) A value of infinity would indicate a spurious alarm and no ground.

B (CORRECT) A ground alarm is received at 10.0KΩ and a reading of zero indicates a severe ground.

C (incorrect) Placing the switch in POS would give a ground alarm but the indication would be +10.0KΩ and the alarm remains locked in until the reset pushbutton is pressed.

D (incorrect) Placing the switch in POS would give a ground alarm but the indication would be +10.0KΩ and the alarm remains locked in until the reset pushbutton is pressed.

Technical Reference(s): SD000188 Rev. 10 Page 12 and Figure 8, 8A and 8BA
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5261 (As available)

Question Source: Bank # _____
Modified Bank # LO001275 (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 H / 2

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
264000 Emergency Generators (Diesel/Jet)	**Tier**	2
	Group	1
A1. Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including:	**K/A #**	264000.A1.03
	Rating	2.8
A1.03 Operating voltages, currents, and temperatures	Rev / Date	2

Proposed Question: RO-22

Columbia is operating in a normal plant configuration for 100% power.

The DG-1 monthly surveillance is in progress and DG-1 is paralleled to SM-7.

A high Drywell Pressure scram occurs.

If all equipment operates as designed, which of the following is correct?

A. DG-1's current increases due to loads starting on SM-7.

B. DG-1's current increases due to DG-1 shifting excitation mode to UNIT.

C. DG-1's current decreases due to no loads now being supplied.

D. DG-1's current decreases due to non-vital loads being load shed.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Plausible as new loads do tie onto SM-7 but incorrect because DG1/7, output breaker, does not re-close onto SM-7. The output breaker would re-close if the LOCA were accompanied by a loss of offsite power.

B (incorrect) Plausible as new loads do tie onto SM-7 but incorrect because DG1/7, output breaker, does not re-close onto SM-7. The output breaker would re-close if the LOCA were accompanied by a loss of offsite power. Plausible because DG-1 will shift to rated speed automatically on an initiation signal but will not shift to UNIT.

C (CORRECT) Current does decrease because DG1/7 opens and does not re-close onto SM-7. DG1/7 would tie back onto SM-7 if there was also a loss of offsite power. The

output breaker would re-close if the LOCA were accompanied by a loss of offsite power.

D (incorrect) Plausible as on SM-7 - SM-75, MC-7C, and MC-7E do load shed. Incorrect because DG1/7 opens and does not re-close onto SM-7. DG1/7 would tie back onto SM-7 if there was also a loss of offsite power. The output breaker would re-close if the LOCA were accompanied by a loss of offsite power.

Technical Reference(s): SD000200 Rev. 012 pages 8, 55, 56,
(Attach if not previously provided, including version/revision number) EWD-47E-003 Rev. 23

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5313 (As available)

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 H / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

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

Proposed Question: RO-23

The plant is operating at 100% power with a monthly DG-2 surveillance in progress. The diesel has been operating at idle speed for five minutes in preparation for going to rated speed. The following events then occur simultaneously:

- A lockout on TR-S (Startup Transformer)
- A lockout on TR-B (Backup Transformer)
- The Main Turbine trips

Which of the following is correct concerning DG-2?

A. The Engine Speed Selector Switch must be placed in RATED, but the output breaker, E-CB-DG2/8, must be manually closed.

B. DG-2 will automatically ramp to rated speed, but the output breaker, E-CB-DG2/8, must be manually closed.

C. DG-2 will automatically ramp to rated speed, and then the output breaker, E-CB-DG2/8, will automatically close.

D. The Engine Speed Selector Switch must be placed in RATED, and then the output breaker, E-CB-DG2/8, will automatically close.

Proposed Answer: C

Explanation (Optional):

A (incorrect) DG-2 will automatic ramp to rated speed. The Engine Speed Selector Switch is procedurally required to be placed in RATED to match the diesel conditions. E-CB-DG2/8 will automatically close due to the undervoltage condition on SM-8.

B (incorrect) The first part of this distractor is correct. The second part is incorrect, because E-CB-DG2/8 will automatically close due to the undervoltage condition on SM-8.

C (CORRECT) DG-2 will automatic ramp to rated speed from idle speed with the Engine Speed Selector Switch in IDLE. Upon reaching rated speed, E-CB-DG2/8 will automatically close due to the undervoltage condition on SM-8.

D (incorrect) Although the Engine Speed Selector Switch is procedurally required to be placed in RATED to match the diesel conditions, it is NOT required for the output breaker to close. This distractor states that after placing the switch in rated, the output breaker will close. This is incorrect because the output breaker will close after the diesel reaches rated speed.

Technical Reference(s): SD000200 Rev. 012 Page 23 and 47
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5321 (As available)

Question Source: Bank # _____
Modified Bank # LO02234 (Note changes or attach parent)
New _____

Question History: Last NRC Exam No Uses
(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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
300000 Instrument Air System (IAS)	**Tier**	2
	Group	1
K1 Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following:	**K/A #**	300000.K1.03
	Rating	2.8
K1.03 Containment air	Rev / Date	2

Proposed Question: RO-24

With Columbia operating at full power, Containment Instrument Air (CIA) pressure begins to lower. ABN-CIA is entered **AND** OPS-2 reports there is a leak on the supply side of the regulator from the normal supply.

Per ABN-CIA as CIA pressure approaches 135 psig, which of the following can be lined up to supply CIA pressure to **ALL** components?

A. Control Air System

B. Service Air

C. Containment Nitrogen

D. Nitrogen Bottle Station

Proposed Answer: A

Explanation (Optional):

A (CORRECT) ABN-CIA directs lining up CAS per SOP-CIA-OPS as pressure approaches 135 psig.

B (incorrect) Service Air does not have a connection to CIA system.

C (incorrect) CN is the normal supply to CIA which is the leak location.

D (incorrect) Nitrogen Bottle Station cannot supply the portion of the system going to the MSIV's.

Technical Reference(s): ABN-CIA Rev. 006 page 4, SD000156 Rev. 011 figure 1
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 F / 2

10 CFR Part 55 Content: 55.41 2

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
300000 Instrument Air System (IAS)	**Tier**	2
	Group	1
K2 Knowledge of electrical power supplies to the following:	**K/A #**	300000.K2.01
	Rating	2.8
K2.01 Instrument air compressor	Rev / Date	2

Proposed Question: RO-25

The power supply to the 'A' Control Air Compressor (CAS-C-1A) is:

A. MC-2P

B. MC-6C

C. MC-7A

D. MC-8A

Proposed Answer: C

Explanation (Optional):

A (incorrect) MC-2P is the power supply to CAS-C-1C.

B (incorrect) MC-6C is the power supply to the Service Air compressor.

C (CORRECT) CAS-C-1A is powered from MC-7A.

D (incorrect) MC-8A is the power supply to CAS-C-1B.

Technical Reference(s): SD000205 Rev. 011 Page 23
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5881 (As available)

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 F / 2

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
400000 Component Cooling Water System (CCWS)	**Tier**	2
	Group	1
	K/A #	400000.A3.01
A3 Ability to monitor automatic operations of the CCWS including:	**Rating**	3.0
A3.01 Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS	Rev / Date	2

Proposed Question: RO-26

The RPV water level trip setpoint for an operating Reactor Closed Cooling water (RCC) pump is…

A. Level 2 on the Narrow Range instrument

B. Level 2 on the Wide Range instrument

C. Level 3 on the Narrow Range instrument

D. Level 3 on the Wide Range instrument

Proposed Answer: B

Explanation (Optional):

A (incorrect) Incorrect but plausible. Level 2 is the trip setpoint for RRC pumps, the HPCS and RCIC initiation setpoint and the NS4 group 2-4 and 7 isolation setpoint and is also the RCC pump trip setpoint. The incorrect part of this distractor is that the trip comes from the Wide range not the Narrow range.

B (CORRECT) This is the RCC pump trip setpoint (NS4 group 4 isolation setpoint) from wide range instruments MS-LS-300A-D.

C (incorrect) Incorrect but plausible. This is the reactor scram setpoint, NS4 Group 5 and 6 isolation setpoint, ADS level confirmation setpoint, and RRC runback setpoint. Additionally, the trip setpoint for RCC pumps comes from the Wide range instruments.

D (incorrect) Incorrect but plausible. Level 3 is the reactor scram setpoint, NS4 Group 5 and 6 isolation setpoint, ADS level confirmation setpoint, and RRC runback setpoint. The choice is plausible as wide range is the correct instrument.

Technical Reference(s): SD000196 Rev 013 Page 15, SD000126 Rev 013 Page 10
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5707 (As available)

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 F / 2

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
201002 Reactor Manual Control System	**Tier**	2
	Group	2
K3. Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following:	**K/A #**	201002.K3.01
	Rating	3.4
K3.01 Ability to move control rods	Rev / Date	2

Proposed Question: RO-27

CRO1 is withdrawing control rods during a reactor startup when the following alarms annunciate:

H13-P603.A7 1-7 RPIS OR DMM INOP
H13-P603.A7 2-8 ROD DRIVE CONTROL SYS INOP

The CRO notes that the control rod being withdrawn has stopped moving and, when attempted, notes another control rod **CANNOT** be selected.

Which of the following would cause these indications?

A. Rod Worth Minimizer (RWM) being inoperable.

B. A transponder card being inoperable.

C. A failure of a Rod Position Information System (RPIS) reed switch.

D. A failure of the Rod Select Matrix pushbutton.

Proposed Answer: B

Explanation (Optional):

A (incorrect) Incorrect but plausible as rod worth minimizer being inoperable would cause a control rod's inability to be moved when below the LPSP, but would not cause a RPIS/DMM alarm or RDCS Inop alarm.

B (CORRECT) A transponder card failure will cause RDCS to fail its self check. The P607 2-8 alarm also causes the 1-7 alarm since RDCS stops sending data to the Display Memory Module (DMM). A different control rod cannot be selected.

C (incorrect) Incorrect but plausible as failure of RPIS reed switch would cause a rod out block alarm but not a RDCS inop alarm.

D (incorrect) Incorrect but plausible - Failure of the Rod Select Matrix would prevent a

different control rod from being selected but does not result in the annunciation. ARP 603.A7 2-8 does not identify this failure as a source of the alarm.

Technical Reference(s): SD000148 Rev. 013 page 20, 21; CGS Simulator;
(Attach if not previously provided, including version/revision number) P603.A7 Rev. 048 drop 1-7 and 2-8

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5787 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
201003 Control Rod and Drive Mechanism	**Tier**	2
	Group	2
K5 Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM:	**K/A #**	201003.K5.04
	Rating	3.1
K5.04 Rod sequence patterns	Rev / Date	2

Proposed Question: RO-28

The plant is shutting down for a planned refueling outage. The following conditions exist:

- Reactor power is 18%
- Rod 46-51 has an insert error and is in the latched group
- Rod 38-31 has an insert error and is in the latched group

The CRO selects the next control rod in the same group, **AND** inserts it two notches past the insert limit.

A control rod…

A. insert error is **NOT** generated until the control rod is inserted an additional notch. Normal control rod insertion may continue until one more insert error is generated.

B. insert error is generated. Normal control rod insertion may continue because rod sequence errors/blocks are **NOT** enforced at this power level.

C. insert block is in effect. Normal control rod insertion is prevented for all rods until a rod with an insert error is bypassed or compliance with the rod sequence pattern is restored.

D. insert block **AND** a withdrawal block is in effect for all rods until a rod with an insert error is bypassed **AND** compliance with the rod sequence pattern is restored.

Proposed Answer: C

Explanation (Optional):

A (incorrect) The first half of the distractor is correct. Rod blocks based on rod sequence are enforced below the Low Power Setpoint (LPSP) of 22%. The stem states reactor power is 18%. This distractor is plausible because the LPSP was previously 17%, which is below the stated power level.

B (incorrect) The first half of the distractor is correct. Rod blocks based on rod sequence are enforced below the Low Power Setpoint (LPSP) of 22%. The stem states reactor power is 18%. This distractor is plausible because the LPSP was previously 17%, which is below the stated power level.

C (CORRECT) An insert block is generated when three insert errors exist. An insert error will occur when a control rod is inserted two notches past the insert limit. Additionally, an insert block will prevent the insertion of all control rods, not just the rods in the latched group. To remove the insert block, one of the rods with an insert error must be bypassed, or a rod must be withdrawn to restore compliance with the rod sequence.

D (incorrect) An insert block is generated when three insert errors exist. An insert error will occur when a control rod is inserted two notches past the insert limit. An insert block will prevent the insertion of all control rods. This is plausible because the insert error will also insert an withdrawal error for all rods except for the three rods with the insert errors. The insert error will also generate a withdrawal block for all rods with the exception of the three rods with the insert errors.

Technical Reference(s): SD000154 Rev.014 pages 8 and 9
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5795 / 5915 (As available)

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 H / 4

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
204000 Reactor Water Cleanup System	**Tier**	2
	Group	2
K1. Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER CLEANUP SYSTEM and the following:	**K/A #**	204000. K1.10
	Rating	3.3
K1.10 Reactor water quality	Rev / Date	2

Proposed Question: RO-29

A plant startup is in progress following a refueling outage, when a RWCU alarm is received. On H13-P602, A5 Drop 2-7 "FILTER DEMIN EFFLUENT CONDUCT HIGH/LOW" is the **ONLY** annunciator in alarm.

Which of the following conditions caused this alarm?

High reactor water conductivity caused by…

A. an end of life resin bed.

B. an overheated resin bed.

C. low resin bed flow.

D. mechanical clogging of the resin bed.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) The FILTER DEMIN EFFLUENT CONDUCT HIGH/LOW alarm comes in at 0.15µS high conductivity. The ARP states this annunciator will come in at the end of a resin beds life.

B (incorrect) Plausible because the overheating of the resin bed could cause damage to the resin resulting in the release of ions into the effluent of the demineralizer resulting in the alarm. If this was caused by the overheating of the resin bed then the CLEANUP FLTR INLET TEMP HIGH alarm drop 6-8 would also be received.

C (incorrect) Plausible because when a demineralizer has been isolated, the effluent conductivity element is in stagnant water and indicated conductivity does rise to

the alarm setpoint. But Vessel A/B Flow Low is an input to the cleanup fltr/demin failure annunciator which would have been received.

D (incorrect) Plausible because the mechanical clogging could cause damage to the resin resulting in the release of ions into the effluent, but not before receiving the CLEANUP FLTR/DEMIN FAUILURE alarm, drop 6-7 on vessel differential pressure.

Technical Reference(s): ARP 4.602.A5 Rev. 043 pages 23, 62, 63;
(Attach if not previously provided, including version/revision number) ARP 4.840.A3 Rev. 019 page 35

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
215001 Traversing In-Core Probe	**Tier**	2
	Group	2
K4 Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following:	**K/A #**	215001.K4.01
	Rating	3.4
K4.01 Primary containment isolation: Mark-I & II (Not-BWR1)	Rev / Date	2

Proposed Question: RO-30

A TIP trace in the core is in progress. The drive control unit is in the Manual Forward Mode **AND** the detector is being driven into the core.

If the reactor were to scram and RPV level were to drop to -55" while the detector is being driven into the core, which of the following is correct?

The TIP detector automatically stops inserting….

A. and withdraws from the core. When the detector reaches the "in-shield" position the ball valve will automatically close.

B. and withdraws from the core. The ball valve must be manually closed when the detector reaches the "in-shield" position.

C. The MANUAL VALVE CONTROL switch must be placed in MANUAL REVERSE position to withdraw the TIP detector. When the detector reaches the "in-shield" position the ball valve will automatically close.

D. The MANUAL VALVE CONTROL switch must be placed in MANUAL REVERSE position to withdraw the TIP detector. The ball valve must be manually closed when the detector reaches the "in-shield" position.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) At -50", TIP receives an NSSS Group 4 isolation signal. The detector stops inserting and immediately withdraws from the core. The ball valve closes when the detector is "in shield".

B (incorrect) The student may think the automatic isolation would not occur and with the TIP trace in progress and may think he has to manually isolate the TIP system.

C (incorrect) The student may think that since the TIP trace is in progress he must retract the TIP Probe before automatic isolation can occur.

D (incorrect) Combination of B and C.

Technical Reference(s): SD000155 Rev. 013 pages 15 and 16
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6989 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode	**Tier**	2
	Group	2
	K/A #	219000.G2.1.7
2.1 Conduct of Operations	**Rating**	4.4
G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	Rev / Date	2

Proposed Question: RO-31

Columbia was operating in MODE 1 with RHR-B in Suppression Pool cooling when an earthquake occurs. As a result, a leak from the Suppression Pool into the RHR-A pump room has been reported. The CRS has entered PPM 5.2.1, Primary Containment Control on low Suppression Pool water level.

The following parameters now exist:

- Suppression Pool temperature on SPTM-TI-5, the digital readout on H13-P601, indicates 115°F.
- Suppression Pool level on CMS-LR-3, wide range, indicates 30' **AND** is trending down slow.

Which of the following is correct?

A. Initiate Standby Liquid Control as the Boron Initiation Temperature has been exceeded.

B. A Reactor scram should be inserted per PPM 5.2.1, Primary Containment Control, due to Suppression Pool temperature exceeding 110°F.

C. Report an EOP entry into PPM 5.2.1, on high Suppression Pool temperature. Maximize Suppression Pool cooling with RHR-P-2B by closing RHR-V-48B.

D. Report the reading on CMS-TR-5(6) Point A02 to the CRS.

Proposed Answer: D

Explanation (Optional):

A (incorrect) This answer is plausible because if Suppression Pool temperature was really was 115°, the BIT temperature would have been exceeded.

B (incorrect) This answer is plausible because PPM 5.2.1, Primary Containment Control directs entering PPM 5.1.1 prior to Suppression Pool temp of 110°F, which then directs initiation of a reactor scram. In this situation the instrument should not be used because of the suppression pool level.

C (incorrect) This answer is plausible as PPM 5.2.1 entry on high Suppression Pool temperature is 90°F and a step directs maximizing Suppression Pool cooling..

D (CORRECT) When Suppression Pool level drops below -6", SPTM-TI-5 becomes uncovered and essentially reads Drywell temperature. There is an operator aid that informs the operator to use CMS-TR-5(6) point A02 if level drops below -6" in the Suppression Pool. No action should be taken based on SPTM-TI-5 when SP/L is LT -6". Control Room staff should now monitor CMS-TR-5(6) pt. A02 instead of SPTM-TI-5 for further evaluation.

Technical Reference(s): PPM 5.2.1 Rev. 22 , PPM 5.1.1 Rev. 20, Operator Aid
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 13278 (As available)

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 H / 2

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
223001 Primary Containment System and Auxiliaries **K6 Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES**: K6.01 Drywell cooling.	**Tier**	2
	Group	2
	K/A #	223001.K6.01
	Rating	3.6
	Rev / Date	2

Proposed Question: RO-32

Columbia is operating at rated power when a spurious trip of CB 7/1 occurs **AND** all systems operate as designed.

Which of the following is correct for the above event?

A. Primary Containment pressure rises. The reactor will scram due to high drywell pressure.

B. Primary Containment pressure rises but stabilizes below the high drywell pressure scram setpoint.

C. The drywell cooling fans will automatically restart when TR-B closes in on SM-7 to limit the rise in Primary Containment temperature/pressure.

D. The immediate operator actions of ABN-ELEC-SM1/SM7 are designed to prevent a reactor scram on high drywell pressure.

Proposed Answer: B

Explanation (Optional):

A (incorrect) A loss of SM-7 causes a loss of half the containment cooling fans. This causes primary containment pressures to rise. Drywell pressure rises but does not cause a reactor scram.

B (CORRECT) A loss of SM-7 causes a loss of half the containment cooling fans. This causes primary containment pressures to rise. Drywell pressure rises but stabilizes at approximately 0.9 psig and does not cause a reactor scram.

C (incorrect) Drywell cooling fans do not automatically restart when SM-7 is repowered from TR-B.

D (incorrect) While ABN-ELEC-SM1/SM7 is entered, there are no immediate operator actions to perform.

Technical Reference(s): CGS Simulator; SD000127 Rev. 015 Pages 11, 12, 31
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11740, 11746 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
233000 Fuel Pool Cooling and Clean-up	**Tier**	2
	Group	2
K2 Knowledge of electrical power supplies to the following:	**K/A #**	233000.K2.02
	Rating	2.8
K2.02 RHR pumps	Rev / Date	2

Proposed Question: RO-33

Which of the following correctly identifies the power supplies to the RHR pumps?

A. RHR-P-2A – SM-8; RHR-P-2B – SM-7; RHR-P-2C – SM-8

B. RHR-P-2A – SM-7; RHR-P-2B – SM-8; RHR-P-2C – SM-8

C. RHR-P-2A – SM-7; RHR-P-2B – SM-8; RHR-P-2C – SM-7

D. RHR-P-2A – SM-7; RHR-P-2B – SM-7; RHR-P-2C – SM-8

Proposed Answer: B

Explanation (Optional):

A (incorrect) RHR-P-2A and RHR-P-2B are reversed.

B (CORRECT) RHR-P-2A is powered from SM-7; RHR-P-2B and RHR-P-2C are powered from SM-8.

C (incorrect) RHR-P-2C is wrong

D (incorrect) RHR-P-2B is wrong

Technical Reference(s): SD000182 Rev. 018 Page 120
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11805 (As available)

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	F / 2
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

Examination Outline Cross-Reference	Level	RO
239001 Main and Reheat Steam System	**Tier**	2
	Group	2
A3 Ability to monitor automatic operations of the MAIN AND REHEAT STEAM SYSTEM including:	**K/A #**	239001.A3.02
	Rating	2.9
A3.02 Opening and closing of drain valves as turbine load changes	Rev / Date	2

Proposed Question: RO-34

A plant startup is in progress with reactor power at 30%. While monitoring indications on H13-P602, CRO-1 observes the Main Steam Line Low Power drain valves, MS-V-69, MS-V-156, and MD-V-73, all indicate ….

A. closed, and must be manually opened when steam flow drops below 5%.

B. closed, and will automatically open when steam flow drops below 5%.

C. open, and will automatically close when steam flow exceeds 50%.

D. open, and must be manually closed when steam flow exceeds 50%.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Plausible because the Main Steam Line Startup Drain Valves (MS-V-16, 19, and 67A-D) are required to be manually shut at 5% power.

B (incorrect) Plausible because the Main Steam Line Startup Drain Valves (MS-V-16, 19, and 67A-D) are required to be manually shut at 5% power, and the Low Power Drains automatically close.

C (CORRECT) The MSL Low power drain valves (MS-V-156, 69 & MD-V-73) automatically close when main steam flow exceeds 50% of rated steam flow.

D (incorrect) Plausible because the Main Steam Line Startup Drain Valves (MS-V-16, 19, and 67A-D) are required to be manually shut, and 50% is the setpoint for auto closure of the Low Power Drain valves.

Technical Reference(s): SD000128 Rev. 012 Pages 18 and 27, EWD-1E-011
(Attach if not previously provided,

including version/revision number) _____

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5534 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
259001 Reactor Feedwater System	**Tier**	2
	Group	2
K1 Knowledge of the physical connections and/or cause – effect relationships between REACTOR FEEDWATER SYSTEM and the following:	**K/A #**	259001.K1.09
	Rating	3.8
K1.09 Reactor water level	Rev / Date	2

Proposed Question: RO-35

Columbia was operating at rated power, when a failure of the FWLC system caused RPV level to increase.

At what RPV water level will the RFW pumps trip?
Considering **ONLY** the RFW system logic, what is the **MINIMUM** number of HIGH LEVEL SEAL IN's that must be reset to restart the RFW pumps?

A. +40.5 inches; 2

B. +54.5 inches; 2

C. +40.5 inches; 3

D. +54.5 inches; 3

Proposed Answer: B

Explanation (Optional):

A (incorrect) +40.5 inches is the High RPV water level alarm setpoint not the RFW pump trip setpoint. At least two of the three seal ins have to reset to allow the HP and LP stops to open on a turbine reset.

B (CORRECT) A RPV level of 54.5" causes both RFT's to trip and a reactor scram at +13 inches due to a loss of feed. At least two of the three seal ins have to reset to allow the HP and LP stops to open on a turbine reset.

C (incorrect) +40.5 inches is the High RPV water level alarm setpoint not the RFW pump trip setpoint. All three seal in's are not required to be reset in order to restart the RFP.

D (incorrect) A RPV level of 54.5" causes both RFT's to trip and a reactor scram at +13 inches due to a loss of feed. All three seal in's are not required to be reset in

order to restart the RFP.

Technical Reference(s): SD000151 Rev. 013 page 27;
(Attach if not previously provided, including version/revision number) SOP-RFT-RESTART-QC Rev. 005 step 2.1.2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5753 (As available)

Question Source: Bank # LO02134
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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
268000 Radwaste	**Tier**	2
	Group	2
A1 Ability to predict and/or monitor changes in parameters associated with operating the RADWASTE controls including:	**K/A #**	268000.A1.01
	Rating	2.7
A1.01 Radiation level	Rev / Date	2

Proposed Question: RO-36

Chemistry requests a batch letdown of water from the RPV for conductivity control. The following conditions exist:

- RWCU pumps are **NOT** available.
- RWCU Demineralizers are **NOT** available.
- 25,000 gallons of water are required to be let down to Radwaste.

Which tank is preferred per SOP-RWCU-OPS to be aligned for let down and why?

A. EDR-TK-2 Waste Collector Tank.
To maintain general area dose rates as low as reasonably achievable.

B. EDR-TK-2 Waste Collector Tank.
The larger tank volume will accommodate the entire amount to be letdown.

C. EDR-TK-5 Waste Surge Tank.
To maintain general area dose rates as low as reasonably achievable.

D. EDR-TK-5 Waste Surge Tank.
The larger tank volume will accommodate the entire amount to be letdown.

Proposed Answer: A

Explanation (Optional):

The candidate should be able to predict that discharging to EDR-TK-2 will minimize Radwaste Building 437' general area dose rates, while discharging to EDR-TK-5 will cause a significant increase in dose rates. Note on page 15 of SOP-RWCU-OPS states that "Unfiltered reactor coolant should not be directed to the Waste Surge Tank (EDR-TK-5) except during emergencies."

A (CORRECT) EDR-TK-2 would be preferred due to the increase potential in dose rates associated with unfiltered letdown. EDR-TK-2 is behind a shielded wall for this reason. The processing of the water in the EDR system can be done in parallel with the letdown by controlling the letdown rate. Also the stem does not state

that it has to be let down in one batch.

B (incorrect) Plausible because this is the correct tank, however this tank has the smaller capacity of the two tanks.

C (incorrect) Plausible because EDR-TK-5 Waste Surge Tank is another tank in the EDR system which can be aligned to accept plant discharge but is not the correct tank. The second half is also incorrect for this tank, EDR-TK-5 is not located behind a shielded wall which will result in increases general area radiation levels.

D (incorrect) Plausible because EDR-TK-5, Waste Surge Tank, is the other tank in the EDR system which can be aligned to accept plant discharge. Also it is the larger of the two tanks, approximately three times larger, that can be lined up but it is not located behind a shielded wall which will result in increases general area radiation levels. The tank capacity for this tank is large enough to accommodate all the water in the stem.

Technical Reference(s): SOP-RWCU-OPS Rev. 010 page 11, & 15,
(Attach if not previously provided, including version/revision number) SD000136 Rev. 011 page 16

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5670 (As available)

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 F / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
286000 Fire Protection System	**Tier**	2
	Group	2
A4 Ability to manually operate and/or monitor in the control room:	**K/A #**	286000.A4.04
	Rating	2.8
A4.04 Fire main pressure	Rev / Date	2

Proposed Question: RO-37

Alarms are received on FCP-1 and FCP-3. During investigation CRO2 notes that fire main pressure is 115 psig and trending down slow.

Which of the following fire pumps should have auto started?

A. Only FP-P-2A

B. Only FP-P-2A and FP-P-2B

C. Only FP-P-2B and FP-P-1

D. Only FP-P-1

Proposed Answer: A

Explanation (Optional):

A (CORRECT) FP-P-2A starts at 120 psig and should have auto started.

B (incorrect) FP-P-2B starts at 110 psig and should not have auto started.

C (incorrect) FP-P-2B starts at 110 psig and should not have auto started.
FP-P-1 starts at 110 psig and should not be running at this pressure.

D (incorrect) FP-P-1 starts at 110 psig and should not be running at this pressure.

Technical Reference(s): SD000177 Rev. 016 Page 27, 28
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5377 (As available)

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 ___F / 2___

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7____

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
290003 Control Room HVAC	**Tier**	2
	Group	2
A2 Ability to (a) predict the impacts of the following on the CONTROL ROOM HVAC; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	**K/A #**	290003.A2.02
	Rating	3.1
A2.02 Extreme environmental conditions	Rev / Date	2

Proposed Question: RO-38

The REMOTE INTAKE DIV 1 RAD HIGH **AND** HI-HI alarms **AND** the REMOTE INTAKE DIV 2 RAD HIGH **AND** HI-HI alarms have annunciated. CRO2 reports the following readings:

WOA-RIS-31A reads 3600 cpm
WOA-RIS-32A reads 3550 cpm
WOA-RIS-31B reads 3000 cpm
WOA-RIS-32B reads 3100 cpm

Which of the following is correct?

A. Both WMA-FN-54A and WMA-FN-54B, Emergency Unit Filter Fans, start. Enter ABN-RAD-CR and direct the NW(#1) Remote Air Intake be isolated.

B. Neither WMA-FN-54A nor WMA-FN-54B, Emergency Unit Filter Fans, start. Enter ABN-RAD-CR and direct the NW(#1) Remote Air Intake be isolated.

C. Both WMA-FN-54A and WMA-FN-54B, Emergency Unit Filter Fans, start. Enter ABN-RAD-CR and direct the SE(#2) Remote Air Intake be isolated.

D. Neither WMA-FN-54A nor WMA-FN-54B, Emergency Unit Filter Fans, start. Enter ABN-RAD-CR and direct the SE(#2) Remote Air Intake be isolated.

Proposed Answer: B

Explanation (Optional):

A (incorrect) WMA-FN-54A/B starts on a high vent rad 'Z' signal but do not start on a remote air intake hi rad signal. ABN-RAD-CR is entered and the highest rad level intake is isolated which is Division 1 which is associated with WOA-RIS-31A and WOA-RIS-32A.

B (CORRECT) WMA-FN-54A/B do not start on a remote air intake hi rad signal. ABN-RAD-CR directs isolating the intake with the highest radiation levels. WOA-RIS-31A and WOA-RIS-32A are associated with the Division 1 remote air intake.

C (incorrect) WMA-FN-54A/B starts on a high vent rad 'Z' signal but does not start on a remote air intake hi-hi rad signal. WOA-RIS-31A and WOA-RIS-32A are associated with the Division 1 remote air intake, not the Division 2 remote air intake.

D (incorrect) WMA-FN-54A/B do not start on a remote air intake hi rad signal. ABN-RAD-CR directs isolating the intake with the highest radiation levels. WOA-RIS-31A and WOA-RIS-32A are associated with the Division 1 remote air intake not the Division 2 remote air intake.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)
ARP 4.826.P1 Rev. 017 drop 2-3; ARP 4.826.P2 Rev. 018 drop 2-2; ABN-RAD-CR Rev. 012 Page 4; SD000201 Rev. 014 Page 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5225 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 5
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295001 Partial or Complete Loss of Forced Core Flow Circulation	**Tier**	1
	Group	1
	K/A #	295001.G2.1.19
G2 Conduct of Operations	**Rating**	3.9
G2.1.19 Ability to use plant computers to evaluate system or component status.	Rev / Date	2

Proposed Question: RO-39

While operating at rated power an event occurs which results in the plant conditions indicated on the TDAS display shown on **figure 1**.

Assuming all automatic actions occurred as expected, and **WITHOUT** operator action, what is the status of the RRC pumps?

A. Both RRC pumps Off.

B. Both RRC pumps operating at 15 Hz.

C. One RRC pump Off **AND** the one RRC pump operating at 60 Hz.

D. One RRC pump operating at 60 Hz **AND** one RRC pump operating at 51 Hz.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Both RRC pumps secured would result in a reactor power of ~ 50% so power indicated in screenshot is too high.

B (incorrect) Both RRC pumps operating at 15Hz would result in a reactor power of ~55% power so power indicated in screenshot is too high.

C (CORRECT) One RRC pump Secured and the other at 60 Hz would result in reactor power at ~70%.

D (incorrect) Loss of a single channel of ASD will cause a runback to 51Hz on the effected RRC pump resulting in a power of ~95% and power is too low for this to be true.

Technical Reference(s): Simulator
(Attach if not previously provided,

including version/revision number) _____

Proposed references to be provided to applicants during examination: Figure 1

Learning Objective: NONE (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 10

55.43 _____

Comments: Figure 1 - Screenshot of TDAS Containment Status Screen 5 minutes after tripping a single RRC pump showing power at approximately 70.

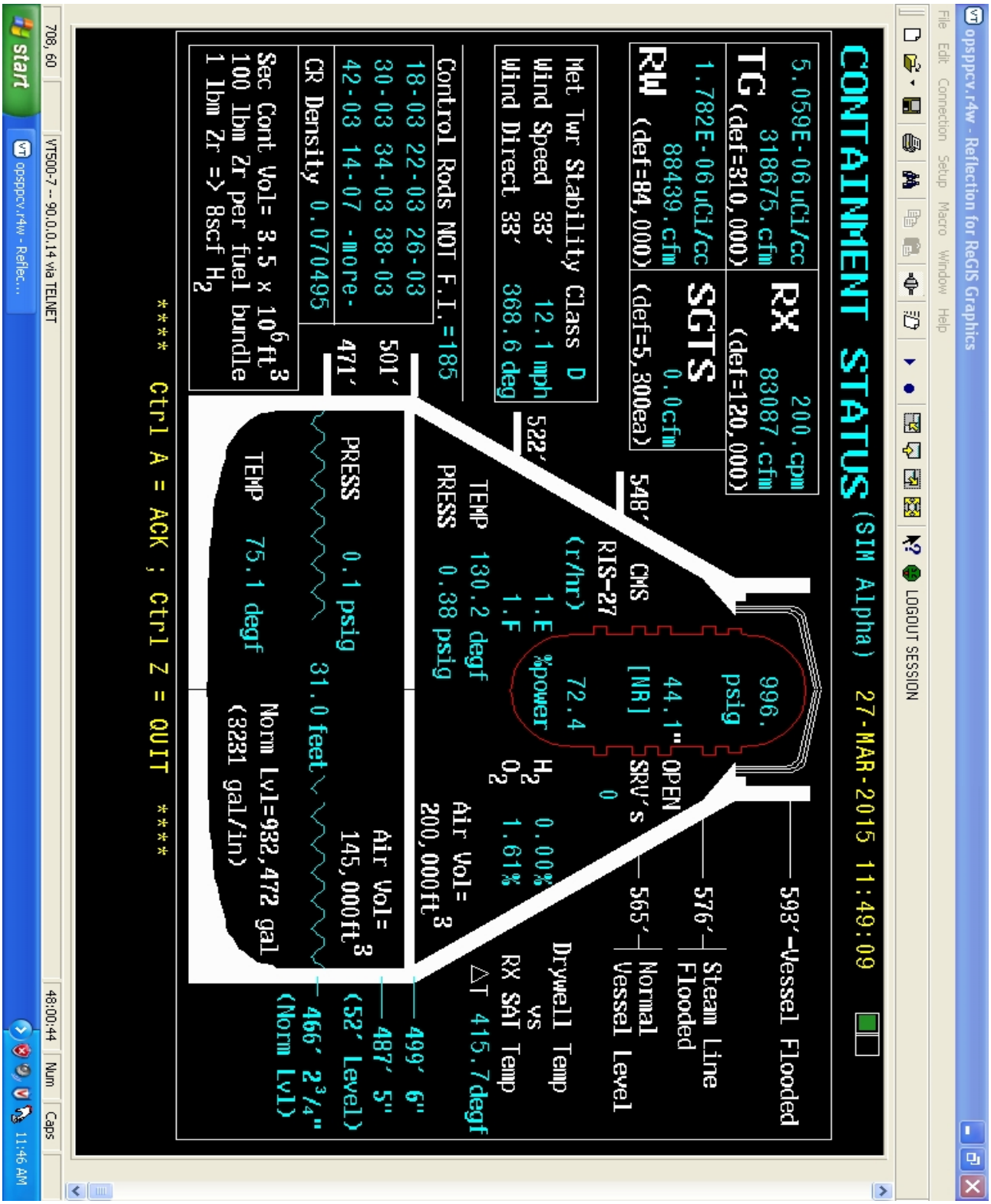

FIGURE 1 (Question RO-39)

Examination Outline Cross-Reference	Level	RO
295003 Partial or Complete Loss of A.C. Power	**Tier**	1
	Group	1
	K/A #	295003.AA2.04
AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER :	**Rating**	3.5
AA2.04 System lineups	Rev / Date	2

Proposed Question: RO-40

Columbia is operating at 100% power when a lockout on SM-2 occurs. All systems operate as designed.

With no operator action, which of the following is the resultant configuration of the feedwater system?

A. RFW-P-1A is running and RFW-P-1B is tripped.

B. RFW-P-1B is running and RFW-P-1A is tripped.

C. Both RFW-P-1A and RFW-P-1B are tripped.

D. Both RFW-P-1A and RFW-P-1B are running.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) A lockout on SM-2 results in a loss of condensate pump 1B and condensate booster pump 2B. This results in low suction pressures for both reactor feed pumps. RFW-P-1B has a four second time delay before it trips and RFW-P-1A has a 10 second time delay. When RFW-P-1B trips, a RRC runback occurs and reactor power is reduced. This power reduction reduces RFW demand which clears the low suction pressure condition and RFW-P-1A continues to operate and restores RPV level.

B (incorrect) If the time delays on the RRW pumps are backwards then this would be the correct answer.

C (incorrect) Would have been true in the past before they staggered the time delays on the RFW pumps.

D (incorrect) If the student thinks that two booster pumps will still provide adequate pressure to the suction of the FRW pumps.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)

ARP 4.840.A1 Rev. 22 drop 8-3 and 8-7;
SD000182 Rev. 018 page 120; Columbia's Simulator

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11576 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295004 Partial or Complete Loss of D.C. Power	**Tier**	1
	Group	1
	K/A #	295004.AA1.02
AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER :	**Rating**	3.8
AA1.02 Systems necessary to assure safe plant shutdown	Rev / Date	2

Proposed Question: RO-41

During operation at power a loss of S1-1 occurs.

If a reactor scram were to occur **AND** the scram valves failed to actuate, which of the following is correct?

CRD-V-110A, backup scram valve, would **NOT** energize, …

A. the amber backup scram lights for Division 1 would energize; the control rods would **NOT** insert.

B. the amber backup scram lights for Division 1 would energize; the control rods would insert.

C. the amber backup scram lights for Division 1 would **NOT** energize; the control rods would **NOT** insert.

D. the amber backup scram lights for Division 1 would **NOT** energize; the control rods would insert.

Proposed Answer: D

Explanation (Optional):

A (incorrect) Incorrect as the Division 1 backup scram lights would not energize on a loss of S1-1. Additionally, the scram air header would still depressurize and rods would insert due to the action of CRD-V-110B.

B (incorrect) Incorrect as the Division 1 backup scram lights would not energize on a loss of S1-1. The second part of this distractor is correct.

C (incorrect) The first part of this distractor is correct – the backup scram lights do not energize. The second part is incorrect because the scram air header would still

depressurize and rods would insert.

D (CORRECT) The amber backup scram lights do not energize on a loss of S1-1 and the header would still depressurize and control rods would insert since CRD-V-110B is actuated by the full scram, is powered from S1-2, and is capable of depressurizing the scram air header.

Technical Reference(s): SD000188 Rev. 010 page 25
(Attach if not previously provided, including version/revision number) _____

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7657 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295005 Main Turbine Generator Trip	**Tier**	1
	Group	1
AK3. Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP:	**K/A #**	295005.AK3.02
	Rating	3.5
AK3.02 Recirculation pump downshift/trip: Plant-Specific	Rev / Date	2

Proposed Question: RO-42

Columbia is operating at rated power at the end of cycle. Events occur that result in a trip of the Main Turbine. CRO1 notes that both Reactor Recirculation Pumps have also tripped.

Which of the following describes the reason the Reactor Recirculation Pumps tripped when the Main Turbine tripped?

A. Prevent cycling SRV's.

B. Prevent power oscillations.

C. Reduce jet pump stall flow.

D. Ensure MCPR is not exceeded.

Proposed Answer: D

Explanation (Optional):

A (incorrect) One or two SRVs do cycle open on a MT trip at rated power.

B (incorrect) Power oscillations occur if operating in certain regions of power to flow map and is not the reason the RRC pumps trip on a MT trip.

C (incorrect) Stall flow occurs when one RRC pump is operating. While tripping both pumps would result in reduced stall flow this is not the reason the RRC pumps trip on a MT trip.

D (CORRECT) EOC RPT mitigates the MCPR vulnerability.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)

SD000129 Rev. 012 Page 39; CGS Simulator;
SD000178 Rev. 016 page 28 and 29

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11647 (As available)

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 F / 2

10 CFR Part 55 Content: 55.41 5

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295006 SCRAM	**Tier**	1
	Group	1
AK2 Knowledge of the interrelations between SCRAM and the following:	**K/A #**	295006 AK2.01
	Rating	4.3
AK2.01 RPS	Rev / Date	2

Proposed Question: RO-43

On panel H13-P610 the following conditions exist:

- The "RPS MG TRANSFER SWITCH" is in NORM.
- The white indicating light associated with "GENERATOR A FEED" is extinguished.
- The white indicating light associated with "ALTERNATE FEED" is extinguished.
- The white indicating light associated with "GENERATOR B FEED" is illuminated.

Based on these conditions:

A. Placing the transfer switch in "ALT A" will result in a full scram.

B. Placing the transfer switch in "ALT B" will result in a full scram.

C. Placing the transfer switch in "ALT A" allows resetting the half scram on RPS A.

D. Placing the transfer switch in "ALT B" allows resetting the half scram on RPS B.

Proposed Answer: B

Explanation (Optional):

A (incorrect) Plausible if the student thinks that the GENERATOR B FEED light indicates that RPS B has lost power. Under these assumption placing RPS MG TRANSFER SWITCH to ALT A would then result in a full scram.

B (CORRECT) The white light off above the GENERATOR A FEED indicates that there is a half scram on RPS system A. Placing the RPS MG TRANSFER SWITCH to the ALT B position will result in a full scram due to a loss of power to RPS B concurrent with the half scram on RPS A.

C (incorrect) Plausible if the student realizes that RPS A has lost power and does not realize that the Alternate source has also lost power. If the Alternate source was available then placing the RPS MG TRANSFER SWITCH to ALT A would energize RPS A, allowing half scram on RPS A to be reset.

D (incorrect) Plausible if the student thinks that the GENERATOR B FEED light indicates that RPS B has lost power. Placing RPS MG TRANSFER SWITCH to ALT B results in a full scram and does not allow resetting the resulting scram on RPS "B".

Technical Reference(s): SD000161 Rev. 017 page 9
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7678 (As available)

Question Source: Bank # LO00865
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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295016 Control Room Abandonment	**Tier**	1
	Group	1
AA2. Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT :	**K/A #**	295016.AA2.06
	Rating	3.3
AA2.06 Cooldown rate	Rev / Date	2

Proposed Question: RO-44

The Control Room was abandoned due to a fire. The Remote and Alternate Remote Shutdown panels have been activated. Actions are underway to place Columbia into Shutdown Cooling. The Control Room Operator is reviewing the Cooldown Temperature/Pressure Data Log which includes the following data:

Time	Temperature (°F)
1000	500
1030	470
1100	430
1130	365
1200	355
1230	295
1300	250
1330	225
1400	175
1430	140
1500	135

Which of the following is the correct concerning this cooldown?

The Technical Specification limit for cooldown rate was……

A. exceeded during **ONLY** one, one hour period.

B. exceeded during **ONLY** two, one hour periods.

C. exceeded during **ONLY** three, one hour periods.

D. exceeded during at least four, one hour periods.

Proposed Answer: B

Explanation (Optional):

A (incorrect) See B

B (CORRECT) Tech Spec cooldown limit is 100°F/Hr and two of the hourly readings exceed that limit (1030 to 1130 and 1200 to 1300). Columbia has an administrative limit of 80°F and that limit was exceed three times (1030 to 1130 and 1200 to 1300 and 1330 to 1430).

C (incorrect) See B

D (incorrect) See B

Technical Reference(s): ABN-CR-EVAC Rev. 033 Page 55
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5005 (As available)

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 H / 2

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	SRO
295018 Partial or Complete Loss of Component Cooling Water	**Tier**	1
	Group	1
	K/A #	295018.AK1.01
AK1. Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER :	**Rating**	3.5
AK1.01 Effects on component/system operations	Rev / Date	2

Proposed Question: RO-45

Columbia is operating at 70% power when a complete loss of Reactor Closed Cooling Water (RCCW) occurs.

A manual scram is directed because of the loss of cooling to the …..

A. Control Rod Drive (CRD) Pump seals.

B. Reactor Recirculation (RRC) Pump seal coolers.

C. Reactor Recirculation (RRC) Pump motors.

D. Reactor Water Clean-Up (RWCU) Non-Regenerative Heat Exchangers.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Is a credible answer as RCC cools the CRD pump seals but not the reason the scram is inserted.

B (incorrect) Is a credible answer as RCC cools the RRC pump seal coolers but is not the reason the scram is inserted. CRD seal cooling provides enough cooling on a loss of RCC.

C (CORRECT) A scram is inserted because the RRC pump motors and seals have lost cooling and they are stopped right after a scram is inserted. (Stopping the RRC P's would require entry into ABN-RRC-LOSS which would also then require a scram).

D (incorrect) Is a credible answer as RCC cools the RWCU NRHXs but is not the reason the

scram is inserted.

Technical Reference(s): ABN-RCC Rev. 006 Page 3, and 7
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 10361 (As available)

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

Question History: Last NRC Exam April 2001
(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 F / 2
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 8
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295019 Partial or Complete Loss of Instrument Air	**Tier**	1
	Group	1
	K/A #	295019.AK2.16
AK2 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following:	**Rating**	2.8
AK2.16 Reactor core isolation cooling	Rev / Date	2

Proposed Question: RO-46

With Columbia operating at full power a total loss of Control Air pressure occurs.

If no operator actions are taken, which of the following **is designed to be utilized in the resultant plant configuration** to maintain RPV level?

A. RCIC-P-1

B. HPCS-P-1

C. CRD-P-1A/1B

D. RFW-P-1A/1B

Proposed Answer: A

Explanation (Optional):

A (CORRECT) Per SD000180, RCIC is designed to be utilized to maintain RPV level in hot shutdown and when the vessel is isolated from the main condenser (MSIVs closed).

B (incorrect) Incorrect but plausible – HPCS could be used as it is a high pressure injection system but per SD000174, HPCS is designed to maintain reactor inventory during a LOCA.

C (incorrect) Incorrect but plausible – The CGS simulator was used and showed that CRD is a viable high pressure injection source that does maintain reactor water level during a loss of CAS where the MSIVs close but is incorrect because it is not designed to maintain RPV level during a transient.

D (incorrect) Incorrect but plausible – The RFW pumps do supply inventory to the RPV but during a complete loss of control air pressure, the MSIVs close, which renders the RFW pumps unavailable.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)

SD000180 Rev. 016 Page 3; SD000174 Rev. 013 Page 3;
FSAR Section 7.4.1.1.1 page 7.4-1,7.4-2 Section 6.3.1.1
page 6.3-1 Section 6.3.2.2.1 page 6.3-7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5713 (As available)

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 H / 2

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295021 Loss of Shutdown Cooling	**Tier**	1
	Group	1
AK3. Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING :	**K/A #**	295021.AK3.04
	Rating	3.3
AK3.04 Maximizing reactor water cleanup flow	Rev / Date	2

Proposed Question: RO-47

Columbia is in a refueling outage with the fuel shuffle underway. A total loss of Shutdown Cooling occurs. The CRS enters ABN-RHR-SDC-LOSS **AND** directs RCC-V-8 (RWCU HX Outlet) to be throttled open.

Why would these actions be taken?

A. To maximize RWCU system flow.

B. To help promote circulation with the spent fuel pool enabling decay heat to be removed by the SFP heat exchangers.

C. To enable natural circulation flow through the reactor providing additional time to recover shutdown cooling.

D. To improve heat transfer rates providing additional time to recover shutdown cooling.

Proposed Answer: D

Explanation (Optional):

A (incorrect) This valve controls RCC flow not RWCU flow. Plausible because of the noun name (RWCU HX Outlet) and the fact that increasing flow through RWCU would increase heat removal.

B (incorrect) Without RHR no flow path exists with the spent fuel pool.

C (incorrect) Natural circulation is promoted by the RPV water level and RWCU flow could possibly disrupt this.

D (CORRECT) Per ABN-RHR-SDC-LOSS bases this improves heat transfer rates providing additional time to recover shutdown cooling.

Technical Reference(s): ABN-RHR-SDC-LOSS Rev. 005 Page 12

(Attach if not previously provided, including version/revision number) _____

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 F / 3

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 5

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295024 High Drywell Pressure	**Tier**	1
	Group	1
EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:	**K/A #**	295024.EA2.04
	Rating	3.9
EA2.04 Suppression chamber pressure: Plant-Specific	Rev / Date	2

Proposed Question: RO-48

A large steam rupture occurs inside containment resulting in a high drywell pressure.

Given:

All automatic actions occur as expected.
No operator actions take place.

Wetwell pressure rises…

A. until it is slightly higher than drywell pressure.

B. remaining 0.2 to 0.5 psi below drywell pressure.

C. until it is approximately 5 psi below drywell pressure.

D. to 6.4 psi above drywell pressure and then equalizes with drywell pressure.

Proposed Answer: C

Explanation (Optional):

A (incorrect) MSIV's will close eventually with no operator action, either on low steam line pressure in RUN or due to high steam tunnel temperature due to load shedding (on high drywell pressure) of the steam tunnel fans, and this may cause SRV operation depending on the size of the rupture. SRV operation alone adds energy to the wetwell and a slow pressurization of the wetwell air space until vacuum breaker operation maintains a d/p of about 0.2 to 0.5 psid. But the dominant effect will be the direct pressurization of the drywell.

B (incorrect) Plausible since the vacuum breakers operate to maintain d/p in this pressure range, but in the opposite direction (relieve the wetwell air space to the drywell rather than the drywell to the wetwell airspace as implied in this proposed response.)

C (CORRECT) The downcomers provide a flow path for uncondensed steam from the drywell to the wetwell during accident conditions. The downcomers are covered by approximately 12 feet of water so the pressure difference will have to overcome the pressure due to the height of that water plus any pressure on the top of the water resulting in approximately 5 psi difference.

D (incorrect) Plausible because 6.4 psid is the design limit for upward pressure on the drywell floor, above this d/p the drywell floor could fail allowing drywell and wetwell air space pressures to equalize. The drywell pressure remains above wetwell pressure when this wetwell pressure is reached following a steam line break inside the drywell.

Technical Reference(s): SD000127 Rev. 0015 pages 8, 22-28, 33, and simulator
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295025 High Reactor Pressure	**Tier**	1
	Group	1
G2 Conduct of Operations	**K/A #**	295025.G2.1.45
	Rating	4.3
G2.1.45 Ability to identify and interpret diverse indications to validate the response of another indication.	Rev / Date	2

Proposed Question: RO-49

While operating at rated power a reactor scram occurs. While verifying plant response after the scram, CRO1 reports that both RRC-P-1A and RRC-P-1B have tripped.

If the Reactor Recirc pumps tripped due to RPV pressure, which of the following is the **MINIMUM** value that RPV pressure reached to cause the pump trips?

A. 1060 psig

B. 1091 psig

C. 1101 psig

D. 1120 psig

Proposed Answer: D

Explanation (Optional):

A (incorrect) At 1060 psig, a reactor scram occurs but the RRC pump would be operating at 15 Hz due to RPV low level.

B (incorrect) 1091 psig is the first setpoint that SRVs open. A reactor scram would have occurred and RRC pumps would be operating at 15 Hz.

C (incorrect) 1101 is the setpoint at which the second set of SRVs would open at. A reactor scram would have occurred and RRC pumps would be operating at 15 Hz.

D (CORRECT) At 1120 psig, the Reactor Recirc pumps trip due to ATWS-ARI signal.

Technical Reference(s): SD000128 Rev. 012 page 9 SD000178 Rev. 016 page 13; SD000126 Rev. 013 page11
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7639 (As available)

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 F / 2

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295026 Suppression Pool High Water Temperature	**Tier**	1
	Group	1
	K/A #	295026.EA1.03
EA1 Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:	**Rating**	3.9
EA1.03 Temperature monitoring	Rev / Date	2

Proposed Question: RO-50

An isolated ATWS has occurred and Suppression Pool temperature is 190°F **AND** rising. RCIC is injecting **AND** has been aligned to take a suction from the Suppression Pool.

Which of the following Suppression Pool temperatures are limits that apply to RCIC?

1. A RCIC NPSH limit is reached when Suppression Pool temperature reaches 240°F
2. A RCIC NPSH limit is reached when Suppression Pool temperature reached 210°F
3. A RCIC lube oil limit is reached when Suppression Pool temperature reached 240°F
4. A RCIC lube oil limit is reached when Suppression Pool temperature reached 210°F

A. 1 and 3

B. 1 and 4

C. 2 and 3

D. 2 and 4

Proposed Answer: C

Explanation (Optional):

A (incorrect) Choice 1 is too high Per PPM 5.2.1 Table 18, the Suppression pool RCIC NPSH temperature limit is 210°F. Choice 3 is correct - Per PPM 5.1.1 Caution 2, the Suppression Pool temperature that may result is high RCIC lube oil temps and bearing temps is 240°F.

B (incorrect) Choice 1 is too high Per PPM 5.2.1 Table 18, the Suppression pool RCIC NPSH temperature limit is 210°F. Choice 4 is too low - Per PPM 5.1.1 Caution 2, the Suppression Pool temperature that may result is high RCIC lube oil temps and bearing temps is 240°F.

C (CORRECT) Per PPM 5.2.1 Table 18, the Suppression pool RCIC NPSH temperature limit is 210°F. Per PPM 5.1.1 Caution 2, the Suppression Pool temperature that may result is high RCIC lube oil temps and bearing temps is 240°F.

D (incorrect) Choice 2 is correct - Per PPM 5.2.1 Table 18, the Suppression pool RCIC NPSH temperature limit is 210°F. Choice 4 is too low Per PPM 5.1.1 Caution 2, the Suppression Pool temperature that may result is high RCIC lube oil temps and bearing temps is 240°F.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)
PPM 5.1.1 Rev. 020 Caution 2,
PPM 5.2.1 Rev. 022 Table 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8498 (As available)

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 F / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295023 Refueling Accidents	Tier	1
	Group	1
2.4 Emergency Procedures / Plan	K/A #	295023.G2.4.18
	Rating	3.3
G2.4.18 Knowledge of the specific bases for EOPs.	Rev / Date	2

Proposed Question: RO-51

A refueling accident occurs resulting in the radioactivity levels at the Exclusion Area Boundary exceeding the Alert Level. PPM 5.4.1 Radioactivity Release Control is entered **AND** the Radwaste Building HVAC is restarted.

Which of the following is the bases for restarting the Radwaste Building HVAC in PPM 5.4.1?

A. Allows for radioactivity to be released at ground level limiting the dispersion of radioactivity.

B. Results in positive pressure in the building to limit intrusion of radioactivity from the reactor building.

C. Prevents Control Room from becoming uninhabitable due to high radiation.

D. Preserves accessibility, and assures that radioactivity is discharged through an elevated, monitored release point.

Proposed Answer: D

Explanation (Optional):
From PPM 5.0.10: Operation of ventilation in this structure preserves accessibility, and assures that radioactivity is discharged through an elevated, monitored release point.

A (incorrect) Radwaste HVAC is an elevated release point, but a ground release would limit dispersion of radioactivity.

B (incorrect) Pressure in the Rad Waste building is slightly negative, but if it were positive it would limit radioactivity from the Reactor Building.

C (incorrect) Control Room is located in the Radwaste building, but it had its own ventilation system separate from the Radwaste building.

D (CORRECT) See note above

Technical Reference(s): PPM 5.0.10 Rev. 19 page 315
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8477 (As available)

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 F / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295028 High Drywell Temperature	**Tier**	1
	Group	1
EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE :	**K/A #**	295028.EK1.02
	Rating	2.9
EK1.02 Equipment environmental qualification	Rev / Date	2

Proposed Question: RO-52

When entering PPM 5.2.1, Secondary Containment Control, Emergency Depressurization is required on High Drywell temperature if it **CANNOT** be restored and maintained below the limit.

This is done to prevent exceeding the design temperature of which of the following components:

A. Drywell Cooling Fans

B. Drywell Recirculation Fans

C. ADS SRV's

D. Drywell to Wetwell Vacuum Breakers

Proposed Answer: C

Explanation (Optional):

A (incorrect) Plausible because the Drywell Cooling Fans are located in the drywell.

B (incorrect) This distractor is plausible since the drywell recirculation fans are credited with mitigating hydrogen buildup and are located in the upper drywell region where air temperature would be the highest.

C (CORRECT) Per PPM 5.0.10 "When drywell temperature cannot be restored and maintained below the ADS design temperature, emergency RPV depressurization is performed."

D (incorrect) Plausible if the student thinks that the high temperature will cause improper operation of the Drywell to Wetwell Vacuum Breakers and causing an overpressurization event upon Emergency Depressurization resulting in damage to the Drywell Floor.

Technical Reference(s): PPM 5.0.10 Rev. 019 page 273, FSAR Section 5.2.2.4
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8318 (As available)

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 F / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 8
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295030 Low Suppression Pool Water Level	**Tier**	1
	Group	1
EA1. Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:	**K/A #**	295030.EA1.03
	Rating	3.4
EA1.03 HPCS: Plant-Specific	Rev / Date	2

Proposed Question: RO-53

What is the PPM 5.2.1, Primary Containment Control, Vortex Limit for the HPCS pump when taking a suction from the Suppression Pool?

A. 17.5'

B. 14.5'

C. 14.0'

D. 5.5'

Proposed Answer: D

Explanation (Optional):

A (incorrect) Suppression Pool Level at 17.5' requires RCIC to be secured if aligned to the Suppression Pool, not HPCS.

B (incorrect) Suppression Pool level at 14.5' require LPCS and RHR-C to be secured – not HPCS.

C (incorrect) Suppression Pool level at 14' requires RHR-A and RHR-B to be secured – not HPCS.

D (CORRECT) A suppression pool level of 5.5' requires HPCS-P-1 to be secured.

Technical Reference(s): PPM 5.0.10 Rev. 019 page 90; EOP 5.2.1 Rev. 22
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8389 (As available)

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 F / 3

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295031 Reactor Low Water Level	**Tier**	1
	Group	1
EK3. Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL :	**K/A #**	295031.EK3.02
	Rating	4.4
EK3.02 Core coverage	Rev / Date	2

Proposed Question: RO-54

During performance of PPM 5.1.1 RPV Control, why is it desired to maintain RPV level greater than -161 inches?

It ensures adequate core cooling through…

A. core submergence.

B. steam cooling with injection.

C. spray cooling with HPCS or LPCS injecting.

D. steam cooling without injection.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) Core submergence is at -161 inches.

B (incorrect) Steam cooling with injection is at -183 inches.

C (incorrect) Spray cooling with HPCS or LPCS is at -210 inches.

D (incorrect) Steam cooling without injection is at -201 inches.

Technical Reference(s): PPM 5.0.10 Rev. 019 page 17
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8018 (As available)

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	F / 2
	Comprehension or Analysis	_____

10 CFR Part 55 Content:	55.41	5
	55.43	_____

Comments:

Examination Outline Cross-Reference	Level	RO
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown	**Tier**	1
	Group	1
	K/A #	295037.EK2.14
EK2. Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following:	**Rating**	3.6
EK2.14 RPIS: Plant-Specific	Rev / Date	2

Proposed Question: RO-55

Which of the following combinations would **NOT** require the Control Room Supervisor to transition from PPM 5.1.1, RPV Control, to PPM 5.1.2, RPV Control - ATWS?

1. 1 rod at RPIS position 48
2. 1 rod at RPIS position 08
3. 1 rod at RPIS position 04
4. 2 rods at RPIS position 02
5. all other rods at RPIS position 04
6. all other rods at RPIS position 02
7. all other rods at RPIS position 00

A. Conditions 1 and 2 and 7

B. Conditions 1 and 4 and 5

C. Conditions 2 and 3 and 6

D. Conditions 2 and 4 and 7

Proposed Answer: D

Explanation (Optional):
This is required Reactor Operator knowledge at Columbia.

A (incorrect) The combination of 1 and 2 makes entry into 5.1.2 required.

B (incorrect) Item 5 makes entry into PPM 5.1.2 required.

C (incorrect) Item 5 makes entry into PPM 5.1.2 required.

D (CORRECT) Per OI 15 the reactor is shutdown with one rod at any position and all others at least inserted to position 02.

Technical Reference(s): OI-15 Rev. 25 Page 12
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7784 (As available)

Question Source: Bank # LO01786
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam March 2009
(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 H / 2
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295038 High Off-Site Release Rate	**Tier**	1
	Group	1
EK1 Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE :	**K/A #**	295038.EK1.02
	Rating	4.2
EK1.02 Protection of the general public	Rev / Date	2

Proposed Question: RO-56

Concerning PPM 5.2.1, Primary Containment Control, what does the following symbol mean?

A. Radioactivity Release; venting is permitted to exceed offsite release rate limits because waiting could result in even greater offsite doses to the public.

B. Radioactivity Release; a release is in progress and actions must be taken to terminate offsite release to the public.

C. Primary Containment Pressure Limit; venting is required to prevent exceeding the pressure limit prior to failure of primary containment resulting in greater offsite doses to the public.

D. Primary Containment Pressure Limit; venting is required to prevent exceeding the pressure limit prior to failure of primary containment, but cannot exceed allowable offsite dose rates.

Proposed Answer: A

Explanation (Optional):
Meaning: Venting may proceed exceeding offsite radioactive release rate limits if necessary.

The consequences of delaying primary containment venting until primary containment failure is imminent could include core damage and even greater offsite doses threatening public health and safety.

A (CORRECT`) Correct Symbol name and meaning as stated above.

B (incorrect) Plausible because correct symbol, but incorrect meaning because an offsite release may not be in progress when the symbol is reached.

C (incorrect) Incorrect symbol, but the symbol for Primary Containment Pressure Limit is used in this EOP flowchart, and it is in the same step as the correct symbol. The actions taken are to prevent exceeding the Primary Containment Pressure Limit which would be done.

D (incorrect) Incorrect symbol, but the symbol for Primary Containment Pressure Limit is used in this EOP flowchart, and it is in the same step as the correct symbol. Actions are taken to prevent exceeding this limit not that the actions will cause you to exceed the limit, but no direction is given to prevent exceeding offsite dose to the public in this step.

Technical Reference(s): 5.0.10 Rev. 019 page 57, 277
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8035 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 8
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
600000 Plant Fire On Site	Tier	1
	Group	1
AK2 Knowledge of the interrelations between PLANT FIRE ON SITE and the following:	K/A #	600000.AK2.01
	Rating	2.6
AK2.01 Sensors / detectors and valves	Rev / Date	2

Proposed Question: RO-57

The plant is operating at power when a spurious fire alarm is locked in on FCP-1 in the Control Room.

Which of the following is correct?

A. A subsequent alarm from that zone will cause a reflash of the fire alarm.

B. A subsequent alarm from that zone will **NOT** annunciate.

C. Subsequent alarms from that zone will annunciate **ONLY** if the locked in alarm is bypassed.

D. Subsequent alarms from that zone will annunciate **ONLY** if the locked in alarm is locally reset.

Proposed Answer: B

Explanation (Optional):

A (incorrect) A spurious alarm prevents any valid alarm in that zone from annunciating. There is no reflash of the annunciator if other detectors in the area subsequently activate.

B (CORRECT) A spurious alarm prevents any valid alarm in that zone from annunciating.

C (incorrect) This distractor is wrong because you can bypass the alarm OR locally reset it in order to allow other detectors to activate an alarm.

D (incorrect) This distractor is wrong because you can bypass the alarm OR locally reset it in order to allow other detectors to activate an alarm.

Technical Reference(s): SD000177 Rev. 016 Page 31
(Attach if not previously provided,

including version/revision number) _____

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7610 (As available)

Question Source: Bank # _____

Modified Bank # LO00586 (Note changes or attach parent)

New _____

Question History: Last NRC Exam NONE

(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 F / 3

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 7

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
700000 Generator Voltage and Electric Grid Disturbances	**Tier**	1
	Group	1
	K/A #	700000.AK3.02
AK3. Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES:	**Rating**	3.6
AK3.02 Actions contained in abnormal operating procedure for voltage and grid disturbances	Rev / Date	2

Proposed Question: RO-58

Columbia has been contacted by the BPA dispatcher and informed that the grid analysis program has projected a degraded reliability of the 230kV power source (E-TR-S) such that post trip voltage for E-TR-S will be less than the allowable limits. The CRS enters ABN-ELEC-GRID.

Which of the following is a required action **AND** explains the reason for the action?

A. Suspend all surveillances which have a potential to trip the Reactor or Main Turbine. This limits the vulnerability of the plant tripping and causing a loss of condensate and feedwater.

B. Declare E-TR-S inoperable and make preparations for a inserting a reactor scram per PPM 3.3.1, Reactor Scram. This places the plant in a safe condition prior to grid instabilities affecting systems required for safe shutdown.

C. Monitor the Generator Exciter Volts DC meter for oscillations. The WTA regulator can not automatically maintain the main generator within the Generator Capability Curve during grid instabilities.

D. Direct a plant shutdown per PPM 3.2.1, Normal Plant Shutdown. This limits the possibility of a Main Turbine/Generator trip and subsequent reactor scram as a result of the potential loss of offsite power.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) Per ABN-ELEC-GRID, surveillances are stopped to limit the vulnerability of the plant tripping which coupled with the loss of the startup transformer would result in a complete loss of condensate and feed.

B (incorrect) Plausible, if conditions are degraded LCO 3.8.1.F may require a plant shutdown which would compound this problem. The ABN states in lieu of performing a plant shutdown a request for discretionary enforcement from the NRC is warranted. PPM 3.2.1 would be used initially to perform the shutdown after 72 hours if enforcement discretion was not received.

C (incorrect) Action is correct per the ABN, however the bases for this step states the power system stabilizer, which is part of the WTA Regulator, is designed to automatically maintain the main generator within the Generator Capability Curve.

D (incorrect) Plausible, conditions may require a plant shutdown which would compound this problem. The ABN states in lieu of performing a plant shutdown a request for discretionary enforcement from the NRC is warranted.

Technical Reference(s): ABN-ELEC-GRID Rev. 007 Pages 5, 6, 12, and13
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 15748 (As available)

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 H / 4

10 CFR Part 55 Content: 55.41 4
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295002 Loss of Main Condenser Vacuum	**Tier**	1
	Group	2
AA2. Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM :	**K/A #**	295002.AA2.02
	Rating	3.2
AA2.02 Reactor power: Plant-Specific	Rev / Date	2

Proposed Question: RO-59

Columbia is starting up with Reactor power currently 20%. A malfunction in the Off Gas system results in increasing main condenser backpressure. The power increase is stopped to investigate.

If main condenser backpressure reaches 6.0 in Hg, which of the following explains the resultant plant response without any further action?

The Main Turbine trips, the Reactor …..

A. scrams due to the Main Turbine trip.

B. scrams due to MSIV closure.

C. remains at power, with a slight increase in power.

D. remains at power, with a slight decrease in power.

Proposed Answer: C

Explanation (Optional):

The Main Turbine trips under all loads at 5.5 in Hg backpressure.

A (incorrect) At 20% power the reactor does not scram due to a MT Trip. (Reactor power GT 30% initiates a scram).

B (incorrect) The MSIVs remain open (don't close until 8.3 in Hg).

C (CORRECT) The MT trip causes a loss of extraction steam to the feedwater heaters which will result in a feedwater temperature decrease which will cause Reactor power to increase.

D (incorrect) Remains at power, but power will increase.

Technical Reference(s): SD000129 Rev. 012 page 31; SD000173 page 7 and 8;
(Attach if not previously provided, including version/revision number) SD000161Rev. 017 page 16

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11645, 11646 (As available)

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 H / 4

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295008 High Reactor Water Level	**Tier**	1
	Group	2
AK2. Knowledge of the interrelations between HIGH REACTOR WATER LEVEL and the following:	**K/A #**	295008.AK2.09
	Rating	3.1
AK2.09 Reactor water cleanup system (ability to drain): Plant-Specific	Rev / Date	2

Proposed Question: RO-60

A plant startup is in progress per PPM 3.1.2, Reactor Startup. RPV level is currently +45" and going up slowly. RPV pressure is steady at 75 psig.

Which of the following could be utilized to reduce RPV level?

A. Let down utilizing the in-service Shutdown Cooling loop.

B. Reduce Control Rod Drive System drive flow.

C. Reduce the demand on the RPV Master Level Controller.

D. Let down utilizing the Reactor Water Cleanup System.

Proposed Answer: D

Explanation (Optional):

A (incorrect) Shutdown Cooling is isolated prior to 48 psig per PPM 3.1.2, Plant Startup.

B (incorrect) While CRD is in service, reducing CRD drive water flow would cause control rod speed to change but does not affect the amount of water going to the RPV.

C (incorrect) The RPV Master Level Controller controls the speed of the RFW pumps at power levels above 20% power. This is plausible because the name of the controller is the Master Level Controller.

D (CORRECT) RWCU is used for letdown at this point in the plant startup.

Technical Reference(s): PPM 3.1.2 Flowchart blocks L4, S9, and S14
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5033 (As available)

Question Source: Bank # LO01120

Modified Bank # _____ (Note changes or attach parent)

New _____

Question History: Last NRC Exam NONE

(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 H / 3

10 CFR Part 55 Content: 55.41 5

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295014 Inadvertent Reactivity Addition	**Tier**	1
	Group	2
2.4 Emergency Procedures / Plan	**K/A #**	295014.G2.4.20
	Rating	3.8
G2.4.20Knowledge of the operational implications of EOP warnings, cautions, and notes.	Rev / Date	2

Proposed Question: RO-61

PPM 5.1.2, RPV Control – ATWS, has a Caution concerning rapid injection.

Which of the following is a consequence of rapid injection of water into the vessel during ATWS conditions per the EOP Bases?

A. Thermal shock to the feedwater nozzles.

B. Fuel damage.

C. Rapid pressure decrease exceeding 100°/hr cooldown limit.

D. Feedpump / main turbine trip on high reactor level.

Proposed Answer: B

Explanation (Optional):

A (incorrect) Incorrect but credible as this caution becomes applicable only after you have stopped and prevented RPV injection therefore condensate and feedwater are not injecting. When RPV level is lowered and feed re-initiated, the feedwater nozzles were uncovered and thermal shock of them is a concern but not the reason for the caution.

B (CORRECT) From PPM 5.0.10 the reason for the caution is to warn the operator of the potential plant response if injection of cold, un-borated water into the core is too rapid. This may result in a large increase in positive reactivity with an attendant reactor power excursion sufficient to substantially damage the core.

C (incorrect) Incorrect but credible as the cooldown rate could be exceeded if injection was too rapid during a low powered ATWS condition.

D (incorrect) Incorrect but credible as the RFP's/ Main Turbine would trip on high RPV water level if the level gets to +54.5" which could occur if rapid injection were to occur.

Technical Reference(s): PPM 5.0.10 Rev. 019 page 151
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8499 (As available)

Question Source: Bank # LR00837
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam NONE
(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 F / 2
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295015 Incomplete SCRAM	**Tier**	1
	Group	2
AA1. Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM :	**K/A #**	295015.AA1.07
	Rating	3.6
AA1.07 Neutron monitoring system	Rev / Date	2

Proposed Question: RO-62

A scram was required and the Mode Switch was placed in SHUTDOWN. APRM downscale lights are illuminated. CRO1 reports 10 control rods still indicate full out.

Perform as immediate actions per PPM 3.3.1, Reactor Scram:

A. Depress the four manual scram pushbuttons.

B. Initiate SLC.

C. Insert SRM/IRMs to monitor reactor power.

D. Initiate ARI.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Per PPM 3.3.1, if the downscale lights are not illuminated the manual scram pushbuttons are not depressed.

B (incorrect) Per PPM 3.3.1, if Reactor power is above 5% then SLC would be initiated.

C (CORRECT) Per PPM 3.3.1, for the conditions listed the SRM and IRM's should be inserted.

D (incorrect) Per PPM 3.3.1, if the downscale lights are not illuminated then ARI would be initiated.

Technical Reference(s): PPM 3.3.1 Rev. 061 Page 7
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6686 (As available)

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 H / 2

10 CFR Part 55 Content: 55.41 7

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295007 High Reactor Pressure	**Tier**	1
	Group	2
AK3.06 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE:	**K/A #**	295007.AK3.06
	Rating	3.7
AK3.06: Reactor/turbine pressure regulating system operation.	Rev / Date	2

Proposed Question: RO-63

Columbia is operating at 100% power with one DEH throttle pressure transmitter out of service.

If one of the remaining DEH pressure transmitter begins slowly drifting, the DEH throttle pressure controller will select the signal from the _____ throttle pressure transmitter to prevent exceeding the _____.

A. higher reading; MCPR Safety Limit

B. lower reading; MCPR Safety Limit

C. higher reading, Steam Dome Pressure Safety Limit

D. lower reading, Steam Dome Pressure Safety Limit

Proposed Answer: A

Explanation (Optional):

A (CORRECT) The higher reading pressure transmitter is selected when the outputs diverge. If a pressure transmitter fails suddenly, the DEH system will shift to using the unaffected instrument but the stem specifies a slow failure. The higher reading pressure transmitter will produce a lower pressure demand signal which results in the governor valves opening and reactor pressure lowering, potentially to the MSIV isolation setpoint of 831 psig. This feature ensures that this failure does not result in an over pressurization transient which would challenge the MCPR limit. If the slow drift was in the low direction and the DEH system selected the lower reading instrument, then reactor pressure would rise; this failure mode is precluded for two transmitter operation by the system design.

B (incorrect) Plausible because the pressure demand signal is opposite of the expected change in reactor pressure. When pressure demand signal increases, the governor valves

open and reactor pressure decreases. The common confusion between pressure transmitter output and pressure demand output has resulted in related questions in the systems phase exam bank having a relatively high miss rate.

C (incorrect) Incorrect because the system design does not prevent exceeding the Steam Dome Pressure Safety Limit but plausible as it a safety limit associated with protecting the reactor vessel and the system designed controller operation does reduce the peak reactor pressure reached during this transient, but MCPR is more limiting.

D (incorrect) Plausible because the pressure demand signal is opposite of the expected change in reactor pressure. When pressure demand signal increases, the governor valves open and reactor pressure decreases. The common confusion between pressure transmitter output and pressure demand output has resulted in related questions in the systems phase exam bank having a relatively high miss rate.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)

SD000146 Rev 10 mr4 page 13-14

FSAR Section 15.2.1.2.3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11660 (As available)

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 H / 3

10 CFR Part 55 Content:

55.41 5

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295032 High Secondary Containment Area Temperature	**Tier**	1
	Group	2
	K/A #	295032.EK1.04
EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE:	**Rating**	3.1
EK1.04 Impact of operating environment on components	Rev / Date	2

Proposed Question: RO-64

A fire exists in the reactor building. ABN-FIRE has been entered.

Concerning operability of RPV water level instruments, which of the following is correct?

A. A fire that has any effect on the operability on one RPV water level instrument requires immediate RRC flow reduction and insertion of a Reactor Scram.

B. A fire in the area of a reference or variable leg for an RPV water level instrument would cause the excess flow check valve for the associated instrument to close, rendering the associated level instrument inoperable.

C. A fire has the potential of causing the reference or variable leg for an RPV water level instrument to heatup, which may cause erroneous indication of RPV water level, rendering associated level instrument inoperable.

D. As long as temperature in the affected reference or variable leg for RPV water level does not exceed the saturation temperature of the water in the RPV, instrument operability need not be questioned.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Plausible if the student thinks that the loss of a single level instrument affects the safe operation of the plant, however this would not be due to the redundancy in the system and would require entry into ABN-INSTRUMENTATION, and the applicable LCO conditions.

B (incorrect) Plausible because the heating up of the reference or variable leg above saturation temperature would cause water flashing to steam resulting in a large pressure

increase. The EFCV is oriented in the wrong direction to stop flow from this direction.

C (CORRECT) Per ABN-FIRE Bases 6.7A fire has the potential of causing the reference or variable legs for the RPV level instrument to heatup. If the temperature of the water in these instruments was to exceed saturation temperature of the water in the RPV, the water inside the instrument leg boils which may cause erroneous indication of RPV level.

D (incorrect) Plausible if the student thinks the only way it would be affected is from flashing of the reference or variable leg due to exceeding saturation conditions, however any change in temperature in either leg could affect the accuracy of the level instrument due to a change in density of the fluid.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)

ABN-FIRE Rev. 034 Page 7, 10, and 30,
SD000126 Rev. 013 page 16

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6904 (As available)

Question Source: Bank # LO01196
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2003 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 H / 3

10 CFR Part 55 Content: 55.41 8
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
295036 Secondary Containment High Sump/Area Water Level	**Tier**	1
	Group	2
	K/A #	295036.EK1.02
EK1. Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL :	**Rating**	2.6
EK1.02 Electrical ground/ circuit malfunction	Rev / Date	2

Proposed Question: RO-65

The "REACTOR BLDG FLOOR SUMP R2 LEVEL HI-HI" annunciator is. All automatic actuations occur as expected but level continues to rise **AND** the water is confined to that room.

If water level continues to rise above the Maximum Safe Operating Levels, grounds could develop on…

A. MC-4A

B. MC-7B

C. SM-7

D. SM-8

Proposed Answer: D

Explanation (Optional):

A (incorrect) Plausible because it is a power supply to equipment in the Reactor Building, but not equipment in the B RHR Pump Room. HPCS Keep Fill Pump power supply.

B (incorrect) Plausible because it is a power supply to equipment in the Reactor Building, but not equipment in the B RHR Pump Room. RCIC Keep Fill Pump power supply.

C (incorrect) Plausible because it is a power supply to equipment in the Reactor Building, but not equipment in the B RHR Pump Room. RHR-A pump

D (CORRECT) The R2 sump is in the B RHR Pump Room and B RHR pump is off SM-8.

Technical Reference(s): ARP 4.6.2.A13 rev. 023 Drop 1-1 page 4;

(Attach if not previously provided, including version/revision number) SD000198 rev. 015 page 48

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 H / 4

10 CFR Part 55 Content: 55.41 8

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.1 Conduct of Operations	**Tier**	3
	Group	N/A
2.1.37 Knowledge of procedures, guidelines, or limitations associated with reactivity management.	**K/A #**	2.1.37
	Rating	4.3
	Rev / Date	2

Proposed Question: RO-66

Columbia is operating a 100% power. An unexpected half scram is received. CRO1 scans the full core display and recognizes that control rod 22-31 has its blue scram light **AND** its green full in light illuminated. Reactor power is observed to have dropped to 98%.

1. Which of the following correctly identifies the 'classification' of control rod 22-31?
2. Which procedure is used to mitigate the resultant consequences?

A. **ONLY** a 'scrammed control rod'.
ABN-ROD.

B. **ONLY** a 'scrammed control rod'.
ABN-POWER.

C. A 'scrammed control rod' **AND** a 'mispositioned control rod'.
ABN-ROD.

D. A 'scrammed control rod' **AND** a 'mispositioned control rod'.
ABN-POWER.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Control Rod is classified as a scrammed control rod and a mispositioned control rod. This choice only indicates the scrammed control rod. ABN-ROD is the correct procedure used to mitigate the resultant consequences.

B (incorrect) Control Rod is classified as a scrammed control rod and a mispositioned control rod. This choice only indicates the scrammed control rod. ABN-POWER is entered but is not used to mitigate the resultant consequences as it directs ABN-ROD be entered.

C (CORRECT) Per SWP-RXE-01, the scrammed control rod is also classified as a mispositioned control rod. ABN-ROD is the procedure used to mitigate the consequences of the scrammed control rod.

D (incorrect) Per SWP-RXE-01, the scrammed control rod is also classified as a mispositioned control rod. ABN-POWER is entered but is not used to mitigate the resultant consequences as it directs ABN-ROD be entered.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)

SWP-RXE-01 Rev. 004 page 25;
ABN-POWER Rev. 013 page 3;
ABN-ROD page 3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6739 (As available)

Question Source: Bank # LO01925
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 F / 2
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 1
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.1 Conduct of Operations	**Tier**	3
	Group	N/A
2.1.29 Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	**K/A #**	2.1.29
	Rating	4.1
	Rev / Date	2

Proposed Question: RO-67

You are sent to complete a valve lineup on the High Pressure Core Spray system per SOP-HPCS-LU. During the performance of the valve lineup you note that the required position for some of the valves is 'C+'.

Which of the following describes the meaning of the 'C+'?

The valve is closed……

A. and torqued to a specific value in the comment section.

B. and logged closed in the control room logs.

C. and a cap is required to be installed.

D. for a valve located in containment.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Torque would be specified in the comments section.

B (incorrect) A comment would be placed in the comments section if it needed to be logged.

C (CORRECT) C+ is a closed valve that has a cap on it.

D (incorrect) +C indicates the valve is in containment.

Technical Reference(s): PPM 1.3.29 Rev. 069 page 8
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 9851 (As available)

Question Source: Bank # LO01839

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 F / 2

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 10

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.1 Conduct of Operations	**Tier**	3
	Group	N/A
2.1.44 Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.	**K/A #**	2.1.44
	Rating	3.9
	Rev / Date	2

Proposed Question: RO-68

During a refueling outage with fuel shuffle in progress, the Control Room Operator is monitoring Source Range Monitors.

If SRM count rate were to increase to exceed 300 counts per second (cps) during a bundle insertion, which of the following is correct?

Immediately stop bundle insertion.

A. Once counts drop to LT 300 cps, insertion can then be continued at a slower rate so as not to exceed 300 cps.

B. Withdraw the bundle from the core and contact the Station Nuclear Engineer for evaluation.

C. Bundle insertion may continue after count rate has stabilized provided SRM count rate does not double on any instrument.

D. Withdraw the bundle until SRM counts drop below 300 cps. Insertion may then continue at a slower rate to prevent exceeding 300 cps.

Proposed Answer: B

Explanation (Optional):

A (incorrect) In the stem it is stated that the count rate exceeds 300 cps which requires the bundle to be withdrawn from the core. Also insertion may not resume until the SNE has been contacted and the cause determined.

B (CORRECT) During refueling a count rate of 300 cps requires the insertion to be immediately stopped and the bundle withdrawn from the core and the SNE notified.

C (incorrect) In the stem it is stated that the count rate exceeds 300 cps which requires the bundle to be withdrawn from the core. Also insertion may not resume until the SNE has been contacted and the cause determined. Plausible because there is a requirement based on doublings which is a criteria for stopping insertion and waiting to observe subcritical multiplication and then you could continue.

D (incorrect) Has the correct withdraw the bundle, but only until counts drop and allows inserting again without contacting SNE

Technical Reference(s):
(Attach if not previously provided, including version/revision number)

PPM 6.3.2
Rev. 23 Page 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8829 (As available)

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 H / 2

10 CFR Part 55 Content: 55.41 10

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.2 Equipment Control	**Tier**	3
	Group	N/A
2.2.1 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	**K/A #**	2.2.1
	Rating	4.5
	Rev / Date	2

Proposed Question: RO-69

While venting Control Rods per PPM 3.1.1 Master Startup Checklist, a Control Rod is encountered that will **NOT** withdraw using normal drive water pressure.

In order to flush air from the insert line what is the **MINIMUM** amount of minutes the control rod should be continuously inserted for?

If this fails what procedure should be used to increase drive water pressure?

A. 1 minute; ABN-ROD

B. 2 minutes; ABN-ROD

C. 1 minute; ABN-CRD

D. 2 minutes; ABN-CRD

Proposed Answer: B

Explanation (Optional):

A (incorrect) Per note on Page 28 of PPM 3.1.1 2 minute insert flush.

B (CORRECT) Per note on Page 28 of PPM 3.1.1 2 minute insert flush. Per page 30 step 8.2.14.m increase drive water pressure per ABN-ROD for those rods which could not be withdrawn at normal drive water pressure.

C (incorrect) Per note on Page 28 of PPM 3.1.1 2 minute insert flush. SOP-CRD-HCU has procedure for filling and venting CRD HCU's but is not the correct procedure called for.

D (incorrect) Per note on Page 28 of PPM 3.1.1 2 minute insert flush. SOP-CRD-HCU has procedure for filling and venting CRD HCU's but is not the correct procedure called for.

Technical Reference(s): PPM 3.1.1 Rev. 055 pages 28-30
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 13301 (As available)

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 F / 2
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 5
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.2 Equipment Control	**Tier**	3
	Group	N/A
2.2.21 Knowledge of pre- and post-maintenance operability requirements.	**K/A #**	2.2.21
	Rating	4.0
	Rev / Date	2

Proposed Question: RO-70

When a safety related motor operated valve has been manually backseated for maintenance, the valve must be declared inoperable until motor operation can be demonstrated.

Select the response below which indicates how the condition of the valve is documented until post maintenance operability requirements are completed.

A. A Caution Tag clearance order shall be issued to document the abnormal condition of the motor operated valve.

B. A Danger Tag clearance order shall be issued to prevent energizing the operator with the valve backseated.

C. A Temporary Modification Request Tag (blue tag) is attached to indicate the condition of the valve.

D. A Maintenance Work Request and its associated Problem Tag is used to document the status of the MOV.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) Per OI-12 Any MOV that is manually backseated should be Caution tagged to document the abnormal condition of this valve.

B (incorrect) This tag is a type of tag used at CGS but not for this purpose.

C (incorrect) This tag is a type of tag used at CGS but not for this purpose.

D (incorrect) This tag is a type of tag used at CGS but not for this purpose.

Technical Reference(s): OI-12 Rev. 041 pg. 41, PPM 1.3.1 Rev. 119 pg. 57
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6259 (As available)

Question Source: Bank # LR01004

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 F / 2

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 10

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.2 Equipment Control	**Tier**	3
	Group	N/A
2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	**K/A #**	2.2.42
	Rating	3.9
	Rev / Date	2

Proposed Question: RO-71

Some instruments in the Control Room have color banding on the indicating range of the meter.

What does operation in the BLUE color banding indicate?

A. Normal Operating Range

B. Alert Range

C. Action Range

D. Technical Specification/LCS/FSAR/ODCM/EOP Requirement Range

Proposed Answer: D

Explanation (Optional):

A (incorrect) This would be a Green Band.

B (incorrect) This would be a Yellow Band.

C (incorrect) This would be a Red Band.

D (CORRECT) Blue banding indicates Technical Specification/LCS/FSAR/ODCM/EOP Requirement, Operation in this range may be a violation of licensing bases.

Technical Reference(s): OI-45 Rev. 6 pg. 4
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 ___F / 3___

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __7__

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.3 Radiation Control	**Tier**	3
	Group	N/A
2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	**K/A #**	2.3.14
	Rating	3.4
	Rev / Date	2

Proposed Question: RO-72

After entering PPM 5.2.1 Primary Containment Control, it is determined that emergency ventilation of the primary containment is required per PPM 5.5.14 Emergency Wetwell Venting.

PPM 5.5.14 directs all personnel in the Reactor Building to be evacuated, what is the bases for this?

A. Venting the Wetwell through the SGT system at high Wetwell pressures may rupture the SGT system ducting and release high energy steam directly into the Reactor Building.

B. Venting the Wetwell through the RB HVAC system at high Wetwell pressures may rupture the RB HVAC system ducting and release high energy steam directly into the Reactor Building.

C. Venting the Wetwell through the SGT system at high Wetwell pressures may rupture the SGT system ducting and release the radioactivity directly into the Reactor Building.

D. Venting the Wetwell through the RB HVAC system at high Wetwell pressures may rupture the RB HVAC system ducting and release the radioactivity directly into the Reactor Building.

Proposed Answer: C

Explanation (Optional):

A (incorrect) Steam entering the Wetwell will have most of its energy removed condensing the steam.

B (incorrect) PPM 5.5.14 directs venting through the SGT system.

C (CORRECT) Caution in PPM 5.5.14: Venting the Wetwell through the SGT system at high Wetwell pressures may rupture the SGT system ducting and release the radioactivity directly into the secondary containment.

D (incorrect) PPM 5.5.14 directs venting through the SGT system.

Technical Reference(s): PPM 5.5.14 Rev. 6 Page 4 and 5
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 7
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.3 Radiation Control	**Tier**	3
	Group	N/A
2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	**K/A #**	2.3.15
	Rating	2.9
	Rev / Date	2

Proposed Question: RO-73

Which of the following is the detector type used by the Main Steam Line (MS-RIS-610A/B/C/D) **AND** the SJAE Condenser Outlet (OG Pretreatment) (OG-RIS-612) radiation monitors?

A. Beta Scintillation detector

B. Geiger Mueller detector

C. Gamma Scintillation detector

D. Ion Chamber detector

Proposed Answer: D

Explanation (Optional):

A (incorrect) See D

B (incorrect) See D

C (incorrect) See D

D (CORRECT) The Main Steam line rad monitor and the OG pre-treatment monitors are Ion Chamber monitors.

Technical Reference(s): SD000147 Rev. 013 page 36
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5646 (As available)

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 ___F / 3___

Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 __12__

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.4 Emergency Procedures / Plan	**Tier**	3
	Group	N/A
2.4.12 Knowledge of general operating crew responsibilities during emergency operations.	**K/A #**	2.4.12
	Rating	4.6
	Rev / Date	2

Proposed Question: RO-74

Initial response stations for the Control Room Operators during abnormal or emergency conditions are:

A. CRO1 responds to H13-P602 and H13-P603 and Board A to control and monitor reactivity and RPV water level; CRO2 responds to H13-P601 to control and monitor ECCS, Containment, and SRV response; the Lead RO responds to Board B and Board C to respond to initial plant concerns.

B. CRO1 responds to H13-P602 and H13-P603 to control and monitor reactivity and RPV water level; CRO2 responds to Board A, Board B, and Board C to respond to initial plant concerns; the Lead RO responds to H13-P601 to control and monitor ECCS, Containment, and SRV response.

C. CRO1 responds to H13-P602 and H13-P603 to control and monitor reactivity and RPV water level; CRO2 responds to H13-P601 responds to control and monitor ECCS, Containment, and SRV response; the Lead RO responds to Board A, Board B, and Board C to respond to initial plant concerns.

D. CRO1 responds to H13-P602, H13-P603, and Board A to control and monitor core reactivity and RPV water level; CRO2 responds to Board B and Board C to respond to initial plant concerns; the Lead RO responds to H13-P601 to control and monitor ECCS, Containment, and SRV response.

Proposed Answer: D

Explanation (Optional):

A (incorrect) This is incorrect because CRO2 responds to Board B and Board C and the Lead RO responds to P601. CRO1 does respond to P602, P603 and Board A.

B (incorrect) This is incorrect because CRO1 also responds to Board A and CRO2 only responds to Board B and Board C. Lead RO does Respond to P601.

C (incorrect) This is incorrect because CRO1 also responds to Board A, CRO2 responds to Board B and Board C and Lead RO responds to P601.

D (CORRECT) This is correct as PPM 1.3.1 requires CRO1 to respond to P602, P603 and Board A, CRO2 to respond to Board B and Board C, and Lead RO to respond to P601.

Technical Reference(s): PPM 1.3.1 Rev.119 page 23
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6088 (As available)

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 F / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	RO
2.4 Emergency Procedures / Plan	**Tier**	3
	Group	N/A
2.4.6 Knowledge of EOP mitigation strategies.	**K/A #**	2.4.6
	Rating	3.7
	Rev / Date	2

Proposed Question: RO-75

The MSIVs close and an ATWS occurs. Actions per PPM 5.1.2, RPV Control – ATWS, are being performed. Level / power conditions exist. Injection systems have been STOPPED **AND** PREVENTED (except SLC, RCIC and CRD Pumps) to lower reactor water level.

Which of the following describes why it is necessary to reduce reactor power by further lowering reactor water level?

A. Limit the energy addition to primary containment.

B. Minimize the amount of fuel damage.

C. Simplify the RPV pressure control with the SRVs.

D. To prevent the possibility of power oscillations occurring.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) RPV level is lowered in an ATWS to reduce reactor power which limits the energy addition to primary containment.

B (incorrect) Plausible as it is something that occurs as a result of lower reactor power but the basis for lowering level is to limit energy addition to PC.

C (incorrect) Plausible as it is something that occurs as a result of lower reactor power but the basis for lowering level is to limit energy addition to PC.

D (incorrect) Plausible, as this is the reason for initially lowering level below -65 inches.

Technical Reference(s): PPM 5.0.10 Rev. 19 Page 137, 145 - 148
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8149 (As available)

Question Source: Bank # LR01022

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 H / 3

10 CFR Part 55 Content: 55.41 10

55.43 _____

Comments:

Examination Outline Cross-Reference	Level	SRO
295004 Partial or Complete Loss of D.C. Power	**Tier**	1
	Group	1
	K/A #	295004 AA2.04
AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER :	**Rating**	3.3
AA2.04 System lineups	Rev / Date	2

Proposed Question: SRO-01 (76)

The MODE SWITCH is in SHUTDOWN and reactor coolant temperature is 250°F, when the feeder breaker from MC-4A to battery charger HPCS-C1-1 trips open.

The **MINIMUM** action(s) required to be taken in order to maintain compliance with the Technical Specifications are contained in Technical Specification(s):

A. 3.8.4, DC Sources Operating, **ONLY**.

B. 3.8.5, DC Sources Shutdown, **ONLY**.

C. 3.8.4, DC Sources Operating, **AND** 3.8.7, Distribution Systems - Operating, **ONLY**.

D. 3.8.5, DC Sources Shutdown, **AND** 3.8.8, Distribution Systems - Shutdown, **ONLY**.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) With the Mode Switch in Shutdown and average reactor coolant temperature greater than 200°F as stated in the stem, the reactor is in Mode 3. LCO 3.8.4, DC Sources – Operating is applicable in Mode 3. The loss of HPCS-C1-1 requires entry into LCO 3.8.4 Condition B (One required Division 3 125VDC battery charger inoperable).

B (incorrect) With the Mode Switch in Shutdown and average reactor coolant temperature greater than 200°F as stated in the stem, the reactor is in Mode 3. Technical Specification 3.8.5 is not entered since it only applies in Modes 4 and 5. Plausible because the LCO does address the operability of the HPCS battery charger.

C (incorrect) With the Mode Switch in Shutdown and average reactor coolant temperature greater than 200°F as stated in the stem, the reactor is in Mode 3. LCO 3.8.4, DC Sources – Operating is applicable in Mode 3. The loss of HPCS-C1-1 requires entry into LCO 3.8.4 Condition B (One required Division 3 125VDC battery

charger inoperable). LCO 3.8.7 is also applicable in Mode 3, but LCO 3.0.6 states, "When a supported system LCO is not met solely due to a support system

LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered." As such, entry into LCO 3.8.7 is not required.

D (incorrect) With the Mode Switch in Shutdown and average reactor coolant temperature greater than 200°F as stated in the stem, the reactor is in Mode 3. Technical Specification 3.8.5 is not entered since it only applies in Modes 4 and 5. Plausible because the LCO does address the operability of the HPCS battery charger. Additionally, LCO 3.8.8 would be the LCO selected if LCO 3.0.6 is incorrectly applied.

Technical Reference(s): TS 3.8.4, 3.8.5, 3.8.7, 3.8.8, 3.5.2, LCO 3.0.6, SD000188,
(Attach if not previously provided, including version/revision number) DC Distribution, Rev.010 pg. 9

Proposed references to be provided to applicants during examination: None

Learning Objective: 7657 (As available)

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

Question History: Last NRC Exam N/A
(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 H3

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments: This is an SRO Only question because in order to answer the question correctly, the applicant must know and apply "motherhood" statement 3.0.6.

Examination Outline Cross-Reference	Level	SRO
295016 Control Room Abandonment	**Tier**	1
	Group	1
AA2. Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT :	**K/A #**	295016 AA2.02
	Rating	4.3
AA2.02 Reactor water level	Rev / Date	2

Proposed Question: SRO-2 (77)

The CRS has entered PPM 5.1.1, RPV Control, and PPM 5.2.1, Primary Containment Control, due to a plant transient. A fire then occurs requiring evacuation of the Main Control Room. After the remote shutdown panel is activated, conditions are as follows:

- RPV Pressure is 1000 psig and up slow.
- Drywell Pressure is 10 psig and up slow.
- Drywell Temperature is 285 F and up slow.
- RPV level is unknown.

Based on these conditions the CRS's highest priority is to…

A. direct the ABN-CR-EVAC Attachments required to isolate and energize SM-8.

B. continue in PPM 5.2.1, Primary Containment Control to address Drywell Temperature.

C. transition to PPM 5.1.4, RPV Flooding, to ensure adequate core cooling.

D. transition to PPM 5.1.3, Emergency RPV Depressurization, in preparation for RPV flooding.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) The unique nature of Control Room Abandonment leads to the unique implementation of the abnormal procedure, ABN-CR-EVAC. Per the procedural note, ABN-CR-EVAC "supersedes EOP procedures and PPM 3.3.1." With the remote shutdown panel activated, the CRS' priority shifts to ensuring SM-8 is isolated and energized in preparation for emergency depressurization due to RPV water level. This will be accomplished by directing performance of the appropriate attachments of ABN-CR-EVAC.

B (incorrect) Per 5.0.10, Flowchart Training Manual, section 3.6, ABN procedures may be

performed but only as a secondary activity. The EOPs have priority/precedence when an EOP entry condition is met. The lone exception to this statement is the implementation of ABN-CR-EVAC, which makes this distractor incorrect. This distractor is plausible because the stem states that Drywell Temperature is 285°F, which exceeds the PPM 5.2.1 threshold for emergency depressurization if temperature cannot be lowered.

C (incorrect) Per 5.0.10, Flowchart Training Manual, section 3.6, ABN procedures may be performed but only as a secondary activity. The EOPs have priority/precedence when an EOP entry condition is met. The lone exception to this statement is the implementation of ABN-CR-EVAC, which makes this distractor incorrect. This distractor is plausible because of the PPM 5.1.1 override which directs the SRO to exit PPM 5.1.1, RPV Control, and transition to PPM 5.1.4, RPV Flooding if RPV level cannot be determined.

D (incorrect) Per 5.0.10, Flowchart Training Manual, section 3.6, ABN procedures may be performed but only as a secondary activity. The EOPs have priority/precedence when an EOP entry condition is met. The lone exception to this statement is the implementation of ABN-CR-EVAC, which makes this distractor incorrect. This distractor is plausible because with RPV level unknown, emergency depressurization is the correct course of action when operating within the EOPs. However, PPM 5.1.4, RPV Flooding contains directions to emergency depressurize the RPV by opening 7 SRV's. In addition, PPM 5.1.1, RPV Control has no direction for the CRS to transition to PPM 5.1.3, Emergency Depressurization.

Technical Reference(s): ABN-CR-EVAC Rev.033 pg. 4 and 7; PPM 5.1.1 Rev.020;
(Attach if not previously provided, including version/revision number) PPM 5.1.4 Rev.010; PPM 5.0.10 Rev.019 pg. 14

Proposed references to be provided to applicants during examination: None

Learning Objective: 6105 (As available)

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

Question History: Last NRC Exam N/A
(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 H2

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments: This is an SRO question because it provides the applicant with a set of plant conditions to assess and requires them to implement an abnormal procedure rather than transitioning to an alternate EOP when override conditions are met.

Examination Outline Cross-Reference	Level	SRO
295019 Partial or Complete Loss of Instrument Air	**Tier**	1
	Group	1
	K/A #	295019 AA2.02
AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR :	**Rating**	3.7
AA2.02 Status of safety-related instrument air system loads (see AK2.1 - AK2.19)	Rev / Date	2

Proposed Question: SRO-03 (78)

With the plant operating in Mode 1, a forklift runs into the Div. 2 ADS Accumulator Backup Compressed Gas Bottle Rack. Current conditions are:

- Four (4) of the Div. 2 bottles are ruptured and depressurized.
- No other bottles or equipment were damaged.
- The other fifteen (15) Div. 2 bottles are at 2400 psig.

Per the Tech Spec Bases, what is the **MINIMUM** number of bottles that must be replaced to return ADS to operable status?

A. 1

B. 2

C. 3

D. 4

Proposed Answer: B

Explanation (Optional):

A (incorrect) Division 2 of ADS contains 19 bottles, 17 of which are required to be operable by Tech Specs. The stem states that 4 bottles have been rendered inoperable. As a result, a minimum of 2 bottles must be replaced to return ADS to operable status. Replacing only 1 bottle would cause ADS to remain inoperable.

B (CORRECT) Division 2 of ADS contains 19 bottles, 17 of which are required to be operable by Tech Specs. The stem states that 4 bottles have been rendered inoperable. As a result, a minimum of 2 bottles must be replaced to return ADS to operable status.

C (incorrect) Division 2 of ADS contains 19 bottles, 17 of which are required to be operable by Tech Specs. The stem states that 4 bottles have been rendered inoperable. As a result, a minimum of 2 bottles must be replaced to return ADS to operable status. Replacing 3 bottles would return ADS to operable status, but does not represent the *minimum* number as the question asks.

D (incorrect) Division 2 of ADS contains 19 bottles, 17 of which are required to be operable by Tech Specs. The stem states that 4 bottles have been rendered inoperable. As a result, a minimum of 2 bottles must be replaced to return ADS to operable status. Replacing 4 bottles would return ADS to operable status, but does not represent the *minimum* number as the question asks.

Technical Reference(s): TS SR 3.5.1.3; TS Bases SR 3.5.1.3; SD000156 Rev.011
(Attach if not previously provided, including version/revision number) pg. 6

Proposed references to be provided to applicants during examination: None

Learning Objective: 5153 (As available)

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

Question History: Last NRC Exam N/A
(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 H3

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments: This question requires knowledge that 17 of 19 bottles for Division 2 ADS are required for operability. This knowledge is TS Bases knowledge.

Examination Outline Cross-Reference	Level	SRO
295023 Refueling Accidents	Tier	1
	Group	1
2.2.40 Ability to apply Technical Specifications for a system.	K/A #	295023 2.2.40
	Rating	4.7
	Rev / Date	2

Proposed Question: SRO-04 (79)

Per the Tech Spec Bases, the **MINIMUM** water level of _____ feet above the top of the RPV flange ensures that sufficient water is available to retain _____ fission product activity in the water in the event of a fuel handling accident.

A. 22; Iodine

B. 22; Krypton

C. 23; Iodine

D. 23; Krypton

Proposed Answer: A

Explanation (Optional):

A (CORRECT) Per LCO 3.9.6, RPV water level shall be ≥ 22 feet above the top of the RPV flange during movement of irradiated fuel within the RPV. The basis for this level is: Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident.

B (incorrect) Per LCO 3.9.6, RPV water level shall be ≥ 22 feet above the top of the RPV flange during movement of irradiated fuel within the RPV. The second part of the distractor is incorrect. The basis for this level is: Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident. Krypton is a plausible distractor because it is a fission product gas that could be released during a fuel handling accident.

C (incorrect) The first part of this distractor is incorrect. Per LCO 3.9.6, RPV water level shall be ≥ 22 feet above the top of the RPV flange during movement of irradiated fuel within the RPV. However, this distractor is particularly plausible because LCO 3.9.7 requires RPV water level to be maintained ≥ 23 feet above the top of irradiated fuel assemblies seated within the RPV. The second part of this distractor is correct. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident.

D (incorrect) The first part of this distractor is incorrect. Per LCO 3.9.6, RPV water level shall be ≥ 22 feet above the top of the RPV flange during movement of irradiated fuel within the RPV. However, this distractor is particularly plausible because LCO 3.9.7 requires RPV water level to be maintained ≥ 23 feet above the top of irradiated fuel assemblies seated within the RPV. The second part of the distractor is incorrect. The basis for this level is: Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident. Krypton is a plausible distractor because it is a fission product gas that could be released during a fuel handling accident.

Technical Reference(s): TS 3.9.6; TS Bases 3.9.6; TS 3.9.7
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: 4169 (As available)

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

Question History: Last NRC Exam N/A
(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 F / 4
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-Reference	Level	SRO
295026 Suppression Pool High Water Temperature	Tier	1
	Group	1
	K/A #	295026 2.2.38
2.2.38 Knowledge of conditions and limitations in the facility license.	Rating	4.5
	Rev / Date	2

Proposed Question: SRO-05 (80)

USE REFERENCES PROVIDED TO ANSWER THIS QUESTION

A plant startup is in progress with the following conditions:

- The Mode Switch is in Startup/Hot Standby
- Suppression Pool average temperature is 93°F and down slow due to the performance of OSP-RCIC/IST-C701, RCIC Post Maintenance Testing
- RCIC testing was completed 3 hours ago
- RCIC is operable and in a standby lineup

Which of the following describes a limitation in the facility license for these conditions?

The Mode Switch…

A. CANNOT be placed in RUN due to Suppression Pool average temperature unless a risk assessment is performed and risk management actions are established.

B. CANNOT be placed in RUN with Suppression Pool average temperature greater than 90°F even if a risk assessment is performed.

C. CAN be placed in RUN without performing a risk assessment until Suppression Pool average temperature exceeds 105°F.

D. CAN be placed in RUN without performing a risk assessment, but Suppression Pool average temperature must be less than or equal to 90°F within 24 hours from now.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) Normally, the mode switch would be allowed to be placed in RUN without performing a risk assessment. In the stem, it states that SP temperature exceeds the limit per LCO 3.6.2.1 (90°F). LCO 3.0.4b permits a Mode change if an LCO is not met after a risk assessment is performed and risk management actions are established.

B (incorrect) The Mode Switch CAN be placed in RUN provided the conditions of LCO 3.0.4b are met. This Spec does not include the "LCO 3.0.4b does not apply" statement.

C (incorrect) The Mode Switch can only be placed in RUN if LCO 3.0.4b is met (risk assessment is performed and risk management actions are established). The RCIC surveillance is no long in-progress, so the 105°F limit is no longer applicable. LCO 3.6.2.1 is not satisfied at 93°F.

D (incorrect) The Mode Switch can only be placed in RUN if LCO 3.0.4b is met (risk assessment is performed and risk management actions are established), but temperature must be restored in 21 hours, not 24.

Technical Reference(s): TS 3.6.2.1; TS Bases 3.6.2.1; LCO 3.0.4
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: TS 3.6.2.1

Learning Objective: 10308 (As available)

Question Source: Bank # 13
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam April 2011 #13
(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 H / 3

10 CFR Part 55 Content: 55.41 _____
55.43 1

Comments:

Examination Outline Cross-Reference	Level	SRO
295038 High Off-Site Release Rate	**Tier**	1
	Group	1
2.4.4 Knowledge of emergency plan protective action recommendations.	**K/A #**	295038 2.4.44
	Rating	4.4
	Rev / Date	2

Proposed Question: SRO-06 (81)

An offsite release is in progress when a field team reports that offsite dose surveys at 1.2 miles indicate a TEDE value of 1450 mrem.

The CRS should declare a …

A. Site Area Emergency and recommend all schools in the EPZ be evacuated.

B. Site Area Emergency and recommend all sectors up to 10 miles be evacuated.

C. General Emergency and recommend all schools in the EPZ be evacuated.

D. General Emergency and recommend all sectors up to 10 miles be evacuated.

Proposed Answer: C

Explanation (Optional):

A (incorrect) The Site Area Emergency threshold for TEDE at 1.2 miles is 100 mrem, and the General Emergency threshold is 1000 mrem. Therefore, a GE should be declared based on the 1450 mrem TEDE value indicated in the stem of the question. The automatic protective action recommendations at the SAE level include evacuating all schools in the EPZ.

B (incorrect) The Site Area Emergency threshold for TEDE at 1.2 miles is 100 mrem, and the General Emergency threshold is 1000 mrem. Therefore, a GE should be declared based on the 1450 mrem TEDE value indicated in the stem of the question. The second part of the distractor is incorrect. Evacuation of all areas in the EPZ is not an automatic SAE PAR. This distractor is plausible because we evacuate 360° at the 2 mile radius at a GE.

C (CORRECT) The General Emergency threshold is 1000 mrem for TEDE at 1.2 miles. Therefore, a GE should be declared based on the 1450 mrem TEDE value indicated in the stem of the question. The second part of the distractor is correct. The automatic protective action recommendations at the SAE level include evacuating all schools in the EPZ.

D (incorrect) The General Emergency threshold is 1000 mrem for TEDE at 1.2 miles. Therefore, a GE should not be declared based on the 125 mrem TEDE value indicated in the stem of the question. The second part of the distractor is incorrect. Evacuation of all areas in the EPZ is not an automatic GE PAR. This distractor is plausible because we evacuate 360° at the 2 mile radius at a GE.

Technical Reference(s): PPM 13.2.2 Rev.018 Attachment 7.1, PPM 13.1.1 Rev.045
(Attach if not previously provided, including version/revision number) Attachment 7.1 Table 4

Proposed references to be provided to applicants during examination: None

Learning Objective: 10189 (As available)

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

Question History: Last NRC Exam N/A
(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 H4

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-Reference	Level	SRO
700000 Generator Voltage and Electric Grid Disturbances	Tier	1
	Group	1
	K/A #	700000 2.2.37
2.2.37 Ability to determine operability and/or availability of safety related equipment.	Rating	4.5
	Rev / Date	2

Proposed Question: SRO-07 (82)

USE REFERENCES PROVIDED TO ANSWER THIS QUESTION

Columbia is operating in Mode 1 with SM-2 being powered from TR-S due to performance of OSP-ELEC-M703 (HPCS DG Monthly Operability Test). DG-3 has been started in preparation for being paralleled to SM-4.

The BPA dispatcher has just notified the CRS that, due to continuing grid disturbances, Startup Transformer voltage is 229 kV and can **NOT** be restored. The CRS updates the crew and enters ABN-ELEC-GRID.

OPS2 then reports steady state voltage for DG3 is reading 3800 volts.

What actions are required per LCO 3.8.1?

A. **ONLY** enter Condition A.
Restore the Startup Transformer to operable status within 72 hours.

B. **ONLY** enter Condition D.
Restore the Startup Transformer or DG-3 to operable status within 12 hours.

C. **ONLY** enter Conditions A and B.
Restore the Startup Transformer or DG-3 to operable status within 72 hours.

D. **ONLY** enter Conditions A, B, and D.
Restore the Startup Transformer or DG-3 to operable status within 12 hours and the other within 72 hours.

Proposed Answer: D

Explanation (Optional):

A (incorrect) If the applicant does not evaluate DG3 as inoperable based on voltage, then only TR-S would be declared inoperable and only Condition A would be entered. This would require TR-S to be restored within 72 hours to comply with the LCO. But because DG3 voltage is reported below the minimum allowable

voltage of 3910 volts, this distractor is incorrect. The TS Bases for LCO 3.8.1 states that the minimum allowable voltage for DG3 allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 3600 V. This makes it plausible that the value stated in the stem (3800 V) would allow DG3 to remain operable since it exceeds the minimum required by the motors to which it provides power.

B (incorrect) With both the Startup Transformer and DG3 inoperable, LCO 3.8.1 Condition D is required to be entered. However, Conditions A and B are still required to be entered and tracked.

C (incorrect) The stem states Startup Transformer (TR-S) voltage is 229 kV, which is below the required voltage of 235.0 kV. This requires TR-S to be declared inoperable, and LCO 3.8.1 Condition A to be entered. DG3 voltage is reported below the minimum allowable voltage of 3910 volts. This requires DG3 to be declared inoperable, and LCO 3.8.1 Condition B to be entered. The combination of the two inoperable power sources requires entry into LCO Condition D.

D (CORRECT) The stem states Startup Transformer (TR-S) voltage is 229 kV, which is below the required voltage of 235.0 kV. This requires TR-S to be declared inoperable, and LCO 3.8.1 Condition A to be entered. DG3 voltage is reported below the minimum allowable voltage of 3910 volts. This requires DG3 to be declared inoperable, and LCO 3.8.1 Condition B to be entered. The combination of the two inoperable power sources requires entry into LCO Condition D. This requires one of the two to be restored within 12 hours. The remaining power source (DG3 or TR-S) must be restored within 72 hours per Condition A/B.

Technical Reference(s): TS 3.8.1; TS Bases 3.8.1; OSP-ELEC-W101 Rev.027 pg. 5
(Attach if not previously provided, including version/revision number) _____

Proposed references to be provided to applicants during examination: TS 3.8.1 (LCO only)

Learning Objective: 3995 (As available)

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 H4

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-Reference	Level	SRO
295015 Incomplete SCRAM	Tier	1
	Group	2
2.1.45 Ability to identify and interpret diverse indications to validate the response of another indication.	K/A #	295015.2.1.45
	Rating	4.3
	Rev / Date	2

Proposed Question: SRO – 08 (83)

A reactor Scram has been initiated. The **ONLY** actions taken were to turn the mode switch to shutdown, and insert the SRM's and IRM's. The Reactor Operators report the following conditions:

- MSIV's are closed.
- RPV level is -25 inches and trending down slow.
- Two SRV's are open.
- RPV pressure is 1030 psig and trending up slow.
- Full Core display is de-energized.
- RWM screen is blank.
- APRM downscale lights are illuminated.
- IRM's are reading 25 on range 10.

Which of the following statements is correct and what action should the CRS direct?

A. APRM downscale lights are correct. Direct lowering reactor pressure to 500-600 psig to allow feeding with condensate booster pumps per SOP-RFW-FCV-QC.

B. APRM downscale lights are correct. Direct starting RCIC and restore level -50 to +54 inches per SOP-RCIC-INJECTION-QC.

C. IRM readings are correct. Direct depressing the manual scram pushbuttons, initiating ARI, and injecting SLC per PPM 5.5.25.

D. IRM readings are correct. Direct depressing the manual scram pushbuttons, initiating ARI, and injecting SLC per SOP-SLC-INJECTION-QC.

Proposed Answer: D

Explanation (Optional):
Two SRVs being open and RPV pressure going up slowly is an indication that an ATWS condition exists and the CRS should direct actions based on this. APRM downscale lights come in at 5% power and lower.

A (incorrect) Incorrect because with two SRV's open and pressure still rising, core thermal power is at least 10%, the APRM downscale lights should not be illuminated. Plausible because unless boron injection reduces power to within the capacity of the combined RCIC, CRD, and SLC injection rates then the EOPs will allow reducing reactor pressure to feed with booster pumps since RFW pumps are not available with the MSIVs closed.

B (incorrect) Level control strategy for ATWS condition would be to lower PRV level to below -65 inches to reduce power.

C (incorrect) PPM 5.5.25 is not directed until the SLC tank is emptied in PPM 5.1.2 but would be directed in PPM 5.1.1 for level control.

D (CORRECT) Correct ATWS conditions exist and SLC should be injected.

Technical Reference(s): PPM 5.0.10 Rev. 19 pg. 185, PPM 5.1.2 RPV Control-ATWS
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-Reference	Level	SRO
295032 High Secondary Containment Area Temperature	**Tier**	1
	Group	2
	K/A #	295032
EA2. Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE :	**Rating**	3.5
	Rev / Date	2
EA2.02 Equipment operability		

Proposed Question: SRO – 09 (84)

USE REFERENCES PROVIDED TO ANSWER THIS QUESTION

The plant is operating at 100% power when the following occurs:

4/1 at 0600: RHR-B Pump Room has exceeded the 150°F limit of LCS 1.7.1C for the past hour
4/3 at 0600: The control power fuses for LPCS-P-1 clear
4/4 at 0600: LPCS-P-1 retesting is complete and LPCS-P-1 is declared operable

If these conditions remain the same, which of the following is required?

A. Enter Mode 3 on 4/8 at 1700.

B. Enter Mode 3 on 4/8 at 1800.

C. Enter Mode 3 on 4/9 at 1800.

D. Increase temperature monitoring to every 4 hours per LCS 1.7.1 **ONLY**.

Proposed Answer: B

Explanation (Optional):
This meets the K/A as operability is tested under high secondary containment temperature as the applicant will need to understand that if the LCS requirement for the RHR pump room temperature is not met, RHR-P-2B must be declared inoperable. This also is at the SRO level of knowledge in that SRO level of knowledge is required to understand that the completion time extension can only be applied if the original inoperability, not the subsequent inoperability is corrected.

A (incorrect) If the candidate thinks RHR-P-2B should have been declared inoperable as soon as the pump room exceeds the LCS 1.7.1 limit, then this would be the correct answer. However, RHR-P-2B is not declared inoperable until the pump room

temperature has exceeded the limit for one hour.

B (CORRECT) The candidate must recognize that exceeding the LCS 1.7.1 Condition C room temperature limit for the past hour requires RHR-P-2B to be declared inoperable on 4/1 at 0600. This requires entry into LCO 3.5.1, Condition A. On 4/3 at 0600, LPCS-P-1 became inoperable. LCO 3.5.1 Condition A is entered again, and Condition C is now entered. On 4/4 at 0600, LPCS-P-1 is restored to operable. This does NOT meet the requirements for completion time extension per TS Section 1.3, because the originally inoperable piece of equipment (RHR-P-2B) remains inoperable. As a result, the original Condition A completion time is applicable (4/8 at 0600). At that time, Condition D must be entered, and Mode 3 entered 12 hours later (4/8 at 1800).

C (incorrect) If the candidate incorrectly applies the completion time extension of 24 hours to the LCO 3.5.1 Condition A completion time, they will arrive at 4/9 at 0600. At that time, Condition D must be entered, and Mode 3 entered 12 hours later (4/9 at 1800).

D (incorrect) This distractor is plausible because the requirements of LCS 1.7.1 Condition B (Perform SR 1.7.1.1 to verify temperatures in the RHR-B Pump Room) remain in effect. The distractor is incorrect because of the additional Tech Spec requirements. The applicant may select this distractor if they do not recognize that the high secondary containment area temperature renders the pump inoperable.

Technical Reference(s): LCS 1.7.1; TS 3.5.1; TS 1.3, Completion Times
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: LCO 3.5.1 and LCS page 1.7.1-1

Learning Objective: NONE (As available)

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 H/4

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-Reference	Level	SRO
295036 Secondary Containment High Sump/Area Water Level	**Tier**	1
	Group	2
	K/A #	295036 EA2.01
EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL :	**Rating**	3.2
EA2.01 Operability of components within the affected area	Rev / Date	2

Proposed Question: SRO-10 (85)

During testing of the Standby Service Water System, the "Reactor Building Equipment Sump R4 Level Hi-Hi" alarm annunciates and clears five minutes later. OPS 2 investigates and reports the RHR-C pump room floor is dry, and there is minimal water in the room's sump. OPS 2 also reports that he was unable to open either door to the LPCS pump room, and that water is seeping from each door's seal. PPM 5.5.27, RB 422 Max Safe Operating Level Measurement, has **NOT** been directed.

The CRS enters PPM 5.3.1, Secondary Containment Control, and ABN-FLOODING.

Based on the above, what actions will the CRS direct?

A. Start RHR-P-2A in SPC and LPCS-P-1 in SP mixing. Both RHR-P-2A and LPCS-P-1 remain operable.

B. Pull the control power fuses for RHR-P-2A and LPCS-P-1. Declare RHR-P-2A and LPCS-P-1 inoperable.

C. Start RHR-P-2A in SPC and pull the control power fuses for LPCS-P-1. Declare both RHR-P-2A and LPCS-P-1 inoperable.

D. Start RHR-P-2A in SPC and pull the control power fuses for LPCS-P-1. **ONLY** declare LPCS-P-1 inoperable.

Proposed Answer: C

Explanation (Optional):

A (incorrect) The first part of the distractor is incorrect. ABN-FLOODING directs starting RHR-P-2A in response to an anticipated loss of LPCS-P-2, and removal of the control power fuses for LPCS-P-1. The distractor is plausible because it is

reasonable to believe that LPCS-P-1 should be started in anticipation of the keepfill pump, LPCS-P-2, just as RHR-P-2A is.

Because prolonged operation in a minimum flow condition is prohibited, the CRS will direct RHR-P-2A to be placed in Suppression Pool Cooling Mode. Due to the possibility of a RHR loop drain down, and subsequent water hammer event following a LOOP concurrently with a LOCA, when RHR is in SPC mode, it is considered inoperable.

B (incorrect) The first part of the distractor is incorrect, because the control power fuses for RHR-P-2A will not be removed. This direction is plausible, because LPCS-P-2 will be stopped due to flooding in the LPCS pump room. This pump supplies keepfill to RHR-P-2A, and a loss of keepfill requires removal of the control power fuses if operation is not required by EOPs.

C (CORRECT) ABN-FLOODING directs starting RHR-P-2A in response to an anticipated loss of LPCS-P-2. Because prolonged operation in a minimum flow condition is prohibited, the CRS will direct RHR-P-2A to be placed in Suppression Pool Cooling Mode. Due to the possibility of a RHR loop drain down, and subsequent water hammer event following a LOOP concurrently with a LOCA, when RHR is in SPC mode, it is considered inoperable.

ABN-FLOODING direct the removal of the control power fuses for LPCS-P-1. This action, by itself, causes LPCS-P-1 to become inoperable.

D (incorrect) The first part of the distractor is correct. ABN-FLOODING directs starting RHR-P-2A in response to an anticipated loss of LPCS-P-2. Because prolonged operation in a minimum flow condition is prohibited, the CRS will direct RHR-P-2A to be placed in Suppression Pool Cooling Mode. Due to the possibility of a RHR loop drain down, and subsequent water hammer event following a LOOP concurrently with a LOCA, when RHR is in SPC mode, it is considered inoperable. This fact makes the second part of the distractor incorrect. This is plausible if the SRO applicant doesn't recall that the act of operating the RHR system in the Tech Spec required SPC mode renders the pump inoperable for LPCI mode.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)
ABN-FLOODING Rev.016 pg.11; SOP-RHR-SPC Rev.008 pg. 6 and 8,; PPM 4.601.A4 Rev.039

Proposed references to be provided to applicants during examination: None

Learning Objective: 10295, 9540 (As available)

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	H3
10 CFR Part 55 Content:	55.41 _____	
	55.43 5	

Comments:

Examination Outline Cross-Reference	Level	SRO
209001 Low Pressure Core Spray System	**Tier**	2
	Group	1
A2. Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	**K/A #**	209001.A2.07
	Rating	2.8
A2.07 Loss of room cooling	Rev / Date	2

Proposed Question: SRO-11 (86)

USE THE REFERENCE PROVIDED TO ANSWER THIS QUESTION

The plant is in Mode 1 with all plant equipment/systems in an Operable status when the following sequence of events take place:

- 4/1 at 1100 One ADS valve is declared inoperable
- 4/5 at 0600 LPCS room cooler is lost

Based on the current situation, Technical Specification 3.5.1 requires the CRS to _____.

A. have the plant in Mode 3 no later than 4/8 at 1800.

B. reduce reactor steam dome pressure to ≤ 150 psig no later than 4/10 at 0600.

C. have the plant in Mode 4 no later than 4/13 at 1800.

D. have the plant in Mode 4 no later than 4/14 at 0600.

Proposed Answer: C

Explanation (Optional):

A (incorrect) The stem states that one ADS valve is inoperable. This leaves 6 operable valves, which meets the requirements of LCO 3.5.1. If this is not recognized, the candidate will enter Condition F and then G. This distractor provides the correct date and time for this condition.

B (incorrect) The stem states that one ADS valve is inoperable. This leaves 6 operable valves, which meets the requirements of LCO 3.5.1. If this is not recognized, the

candidate will enter Condition F and then G. If the candidate then applies the 36 hours to reduce pressure to the time required to be in Mode 3, they will arrive at the date and time given in this distractor.

C (CORRECT) The stem states that one ADS valve is inoperable. This leaves 6 operable valves, which meets the requirements of LCO 3.5.1. LPCS room cooling is required for pump operability. Therefore, when room cooling is lost, TS 3.5.1 Condition A is entered. After 7 days (4/12 at 0600), Condition D is entered. The plant is required to be in Mode 4 36 hours later (4/13 at 1800).

D (incorrect) The stem states that one ADS valve is inoperable. This leaves 6 operable valves, which meets the requirements of LCO 3.5.1. LPCS room cooling is required for pump operability. Therefore, when room cooling is lost, TS 3.5.1 Condition A is entered. After 7 days (4/12 at 0600), Condition D is entered. The plant is required to be in Mode 4 36 hours later (4/13 at 1800). If the candidate applies the 36 hours to reach Mode 4 to the Mode 3 completion time, they will arrive at the date and time given in this distractor.

Technical Reference(s): TS 3.5.1 and TS 3.5.1 Bases; OI-41 Rev.056, page 51
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: TS 3.5.1

Learning Objective: NONE (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 _____
55.43 2

Comments:

Examination Outline Cross-Reference	Level	SRO
209002 High Pressure Core Spray System (HPCS)	Tier	2
	Group	1
	K/A #	209001.2.2.12
2.2.12 Knowledge of surveillance procedures.	Rating	4.1
	Rev / Date	2

Proposed Question: SRO – 12 (87)

While operating in Mode 1, the HPCS Fill Verification surveillance, OSP-HPCS-M101, is in progress. During the surveillance, the following events occur:

0900 – HPCS-V-87 (Injection Line Vent Isolation) is opened.
0920 – HPCS-V-88 (Injection Line Vent Isolation) is opened 1/8 turn.
0922 – A steady stream of water is observed from HPCS-V-88. The elapsed time for venting from HPCS-V-88 is determined to be 75 seconds.
0923 – HPCS-V-88 is closed.

1) Per OSP-HPCS-M101, what is the latest time that HPCS-P-1 can still be considered operable?
2) In accordance with the technical specification bases for TS 3.5.1, this surveillance is conducted in order to_____.

A. 0919; prevent pump cavitation following an ECCS initiation signal.

B. 0922; prevent pump cavitation following an ECCS initiation signal.

C. 0919; prevent water hammer following an ECCS initiation signal.

D. 0922; prevent water hammer following an ECCS initiation signal.

Proposed Answer: C

Explanation (Optional):

A (incorrect) First part is correct. Second part is incorrect. Pump cavitation is not included in the TS basis and would not be a concern for air voids in the pump discharge piping. Plausible if the applicant thinks that the fill verification is conducted on the suction piping.

B (incorrect) First part is incorrect. Pump is considered inoperable when V-88 is opened. Plausible if applicant does not know that an open V-88 takes the pump out of service and determines that the pump is inoperable due to venting time greater than 10.6 seconds - or - if applicant determines that the pump is inoperable after

V-87 is opened and considers the pump to be operable once a steady stream of water is identified. Second part is incorrect (see distractor A).

C (CORRECT) TS 3.5.1 is entered prior to opening V-88. The basis for the surveillance includes preventing a water hammer event on ECCS actuation. SRO justification – applicant must have knowledge of the TS basis document to correctly answer the question.

D (incorrect) First part is incorrect (see distractor B). Second part is correct.

Technical Reference(s): OSP-HPCS-M101 Rev. 007, TS Bases 3.5.1
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5432 (As available)

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 F/2

10 CFR Part 55 Content: 55.41 _____
55.43 13

Comments:

Examination Outline Cross-Reference	Level	SRO
217000 Reactor Core Isolation Cooling System (RCIC)	**Tier**	2
	Group	1
	K/A #	217000 A2.15
A2. Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	**Rating**	3.8
A2.15 Steam line break	Rev / Date	2

Proposed Question: SRO-13 (88)

Columbia is operating at 100% power, when the following occurs:

- An unisolable break develops in the steam supply line to RCIC.
- The Division 1 "LEAK DET RCIC PIPE ROUTING AREA TEMP HI-HI" annunciator on H13-P601-A3 is in alarm.
- The Division 2 "LEAK DET RCIC PIPE ROUTING AREA TEMP HI-HI" annunciator on H13-P601-A2 is **NOT** in alarm.

The CRS enters PPM 5.3.1, Secondary Containment Control as a result.

1) What is the status of RCIC-V-1 (RCIC Turbine Trip MO Valve) now?
2) If temperature in the RCIC pipe routing area then increases to 220°F, the CRS will transition to _____ from PPM 5.3.1.

A. 1) RCIC-V-1 is closed.
   2) PPM 3.2.1, Normal Plant Shutdown.

B. 1) RCIC-V-1 is closed.
   2) PPM 5.1.1 RPV Control.

C. 1) RCIC-V-1 is open.
   2) PPM 3.2.1 Normal Plant Shutdown.

D. 1) RCIC-V-1 is open.
   2) PPM 5.1.1 RPV Control.

Proposed Answer: B

Explanation (Optional):

A (incorrect) 1) Correct - A RCIC turbine trip is regarded as anything that will shut RCIC-V-1. On the Division 1 isolation signal, RCIC-V-1 and RCIC-V-8 close per ARP 4.601.A3 drop 1-4.

2) Incorrect – PPM 3.2.1 would be the correct procedure if the maximum safe operating conditions were met in two or more areas without the primary system discharging into the secondary containment.

B (CORRECT) 1) Correct - A RCIC turbine trip is regarded as anything that will shut RCIC-V-1. On the Division 1 isolation signal, RCIC-V-1 and RCIC-V-8 close per ARP 4.601.A3 drop 1-4.

2) Correct – With the alarm stated in the stem coming in at 160°F if the isolations had properly occurred then the temperature could not have continued to rise to the Maximum Safe Operating Value of 220°F. This means that the Primary System must still be leaking into the Secondary Containment. PPM 5.1.1 is transitioned into via PPM 5.3.1 even though there are not direct entry conditions for PPM 5.1.1.

C (incorrect) 1) Incorrect – This would be correct if it were the Division 2 alarm that had been received instead of the Division 1.

2) Incorrect – PPM 3.2.1 would be the correct procedure if the maximum safe operating conditions were met in two or more areas without the primary system discharging into the secondary containment.

D (incorrect) 1) Incorrect – This would be correct if it were the Division 2 alarm that had been received instead of the Division 1.

2) Incorrect – PPM 3.2.1 would be the correct procedure if the maximum safe operating conditions were met in two or more areas without the primary system discharging into the secondary containment.

Technical Reference(s): PPM 4.601.A3 Rev.025 pg.9; PPM 4.601.A2 Rev.026 pg.6; SD000180 Rev. 16 pg. 25, and 26
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11675, 6906, 5722 (As available)

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	H / 3
10 CFR Part 55 Content:	55.41 _____	
	55.43 5	

Comments:

Examination Outline Cross-Reference	Level	SRO
264000 Emergency Generators (Diesel/Jet)	**Tier**	2
	Group	1
2.2.17 Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.	**K/A #**	264000 2.2.17
	Rating	3.8
	Rev / Date	2

Proposed Question: SRO-14 (89)

While operating at 100% power, an air leak is discovered on the air start motor for DG3. A maintenance worker asks for permission to perform work on the air leak.

1) The CRS should use _____ to determine the overall increase core damage risk for this evolution prior to authorizing maintenance.

While maintenance is being performed with DG3 out of service, it is determined that one offsite AC power source is no longer operable or available.

2) Per the Tech Spec Bases, this condition is a _____ severe degradation in AC power sources than a loss of both offsite AC power sources with all DGs in an operable status.

A. 1) OI-68, Operational Aggregate Risk Assessment Process
2) less

B. 1) OI-68, Operational Aggregate Risk Assessment Process
2) more

C. 1) the Paragon program
2) less

D. 1) the Paragon program
2) more

Proposed Answer: D

Explanation (Optional):

A (incorrect) 1) Incorrect - OI-68 is plausible because it was used in the past, but within the last year this procedure was removed from use.

2) Incorrect - The condition is more severe per TS bases section B3.8.1.D.1 &

D.2.

B (incorrect) 1) Incorrect - OI-68 is plausible because it was used in the past, but within the last year this procedure was removed from use.

2) Correct - The condition is more severe per TS bases section B3.8.1.D.1 & D.2.

C (incorrect) 1) Correct - Per PPM 1.5.14 we use the Paragon Program in Assessing Risk.

2) Incorrect - The condition is more severe per TS bases section B3.8.1.D.1 & D.2.

D (CORRECT) 1) Correct - Per PPM 1.5.14 we use the Paragon Program in Assessing Risk.

2) Correct - The condition is more severe per TS bases section B3.8.1.D.1 & D.2.

Technical Reference(s): PPM 1.5.14 Rev. 036 page 4; TS Bases 3.8.1.D1, and D2
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 13394 (As available)

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 H / 2

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

Examination Outline Cross-Reference	Level	SRO
262002 Uninterruptable Power Supply (A.C./D.C.)	**Tier**	2
	Group	1
	K/A #	262002 A2.02
A2. Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	**Rating**	2.7
A2.02 Over voltage	Rev / Date	2

Proposed Question: SRO-15 (90)

The Alternate Power Supply to IN-1 is tagged out for maintenance.
An event causing the Normal Power Supply voltage to IN-1 to be 106% of the normal voltage occurs.

What is the immediate impact to US-PP?
What will the CRS direct in this situation?

A. US-PP is deenergized.
Enter ABN-POWER and reduce RRC flow.

B. US-PP remains energized.
When the Overvoltage condition clears, recover the inverter per SOP-ELEC-IN1-OPS.

C. US-PP is deenergized.
Enter ABN-FWH-TRIP/HIGH to recover the feedwater heaters.

D. US-PP remains energized.
Transfer the inverter to the Bypass Source using the Kirk Key Interlock per SOP-ELEC-IN1-OPS.

Proposed Answer: A

Explanation (Optional):
RO knowledge is to know the impact on the UPS unit to overvoltage conditions. Static Switch for IN-1 will transfer to the alternate source if it senses an overvoltage condition set at 105% of the normal voltage. IN-2 and 3 do not have this transfer feature. With the alternate power supply tagged out, this automatic transfer results in a loss of power to US-PP.

SRO only knowledge is required to correct the condition using procedures.
SOP-ELEC-IN1-OPS have the procedural steps to transfer to the DC power supply and to transfer using the Kirk Key to the Bypass source. The SRO knowledge is to know that the 105% voltage transfer is to protect the Inverter not US-PP.
The loss of US-PP causes a partial loss of feedwater heating requiring an entry into ABN-POWER.

A (CORRECT) See above

B (incorrect) Plausible because the Maintenance Bypass source is available; but this requires a manual transfer so US-PP does not remain energized.

C (incorrect) Plausible because the loss of power to US-PP results in the feedwater heater controllers losing power. The associated feedwater heater level control valves fail open resulting in low feedwater heater levels, not high level trips.

D (incorrect) Plausible because other inverters (IN-2 and IN-3) do not have the auto transfer feature on over voltage. In this case manually transferring US-PP to another power supply to correct the high voltage condition would be credible.

Technical Reference(s): SD000194 rev. 11 pg. 8, SOP-ELEC-IN1-OPS rev. 003
(Attach if not previously provided, including version/revision number) pg. 11-13; ABN-ELEC-INV rev. 012 pg. 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: NONE (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 5
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	SRO
202001 Recirculation System	**Tier**	2
	Group	2
A2. Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	**K/A #**	202001.A2.05
	Rating	4.0
A2.05 Inadvertent recirculation flow increase	Rev / Date	2

Proposed Question: SRO – 16 (91)

The CRS directs power to be raised with flow from 86% to 100% following economic dispatch.

Initial conditions are:

Master Controller (RRC-M/A-R675) indicates 46 Hz
Loop A and B Drive Flow (RRC-FI-R676A/B) indicates 33,000 GPM
Individual Loop 'A' and 'B' Controllers (RRC-M/A-R676A and RRC-M/A-676B) are in AUTO indicating:

- 46 Hz Actual
- 46 Hz Demand
- 0 Hz Deviation / Bias

When raising flow, the Individual Loop 'B' Controller raise pushbutton is inadvertently depressed and the pushbutton sticks in the depressed position. The Reactor Operator does **NOT** identify the problem until after RRC-P-1B's speed stops increasing.

What effect does the stuck pushbutton have on the RRC system, and what action will the CRS direct to mitigate the event?

Loop A Drive Flow remains the same and Loop B Drive Flow will increase …

A. until the Deviation / Bias meter indicates 6 Hz and then stop resulting in a Loop B Drive Flow of approximately 37,000 GPM.
The CRS will direct stopping B RRC pump and enter ABN-RRC-LOSS.

B. for 15 seconds resulting in a Loop B Drive Flow of approximately 44,000 GPM.
The CRS will direct stopping B RRC pump and enter ABN-RRC-LOSS.

C. until the Deviation / Bias meter indicates 6 Hz and then stop resulting in a Loop B Drive Flow of approximately 37,000 GPM.
The CRS will direct lowering B RRC pump speed to 46 Hz in accordance with ABN-POWER.

D. for 15 seconds resulting in a Loop B Drive Flow of approximately 44,000 GPM.
The CRS will direct lowering B RRC pump speed to 46 Hz in accordance with ABN-POWER.

Proposed Answer: C

Explanation (Optional):

A (incorrect) The first part of this distractor is correct. With the flow controllers in AUTO, the Deviation / Bias will be limited to 6 Hz and stop the increase in flow. Because the rise in pump speed is stopped and is now under control, securing RRC-P-1B it is not required. If stopping RRC-P-1B were still required, entry into ABN-RRC-LOSS would also be required.

B (incorrect) The first part of the distractor is incorrect, because the flow increase will automatically terminate when a 6 Hz mismatch occurs. The distractor is plausible, because if Individual Loop Controller was in Manual, then the flow increase would have continue for 15 seconds (Stuck Pushbutton Timer), resulting in a frequency of approximately 60.5 Hz and 44,000 GPM flow. Because the rise in pump speed is stopped and is now under control, securing RRC-P-1B it is not required. If stopping RRC-P-1B were still required, entry into ABN-RRC-LOSS would also be required.

C (CORRECT) With the flow controllers in AUTO, the Deviation / Bias will be limited to 6 Hz and stop the increase in flow. Given that RRC flow has risen and RRC system flow control is restored, ABN-POWER directs reducing the RRC flow to the pre-transient value.

D (incorrect) The first part of the distractor is incorrect, because the flow increase will automatically terminate when a 6 Hz mismatch occurs. The distractor is plausible, because if Individual Loop Controller was in Manual, then the flow increase would have continue for 15 seconds (Stuck Pushbutton Timer), resulting in a frequency of approximately 60.5 Hz and 44,000 GPM flow. Given that RRC flow has risen and RRC system flow control is restored, ABN-POWER directs reducing the RRC flow to the pre-transient value.

Technical Reference(s):
(Attach if not previously provided, including version/revision number)

Simulator; SD000184 Rev.019 pages 10-13; ABN-POWER Rev.013 pg. 4-5

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 11788 (As available)

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 H / 4

10 CFR Part 55 Content: 55.41 _____

55.43 5

Comments:

Examination Outline Cross-Reference	Level	SRO
226001 RHR/LPCI: Containment Spray System Mode	**Tier**	2
	Group	2
	K/A #	226001.2.2.40
2.2.40 Ability to apply Technical Specifications for a system.	**Rating**	4.7
	Rev / Date	2

Proposed Question: SRO – 17 (92)

USE REFERENCES PROVIDED TO ANSWER THIS QUESTION

Columbia is operating in Mode 1 when the following occurs:

1st of month at 0900 - RHR-P-2A failed to start during the RHR Loop A Operability Test (OSP-RHR/1ST-Q702).

5th of month at 1000 - RHR-V-17B (RHR outboard Drywell spray) is declared INOP and is deactivated in the closed position due to having a wrong sized motor installed.

5th of month at 1600 - The cause of RHR-P-2A's failure to start is corrected, and after testing, is declared operable.

What is the longest time that RHR-V-17B can remain INOP before Columbia must be in **MODE 3**?

A. 8th of the month at 2100

B. 9th of the month at 2100

C. 12th of the month at 2200

D. 13th of the month at 2200

Proposed Answer: B

Explanation (Optional):

A (incorrect) This would be correct if 0900 were used as start point and the 24 hour extension was not applied as allowed by TS 1.3 Completion Times.

B (CORRECT) TS 3.6.1.5.A is entered for RHR A at time 0900 on the 1st. TS 3.6.1.5.A for RHR B is entered at 1000 on the 5th and TS 3.6.1.5.B is entered at 1000 on the 5th for loss of both spray systems. When RHR A is restored, TS 3.6.1.5.B is exited, but TS 3.6.1.5A remains in effect from original inop time (0900 on the

1st). Per TS 1.3 Completion Times, an additional 24 hours may be added to the completion time making the completion time 8 days, and add in 12 hours to be in mode 3 the answer is 2100 on the 9th of the month.

C (incorrect) If the time RHR-V-17B becomes inoperable is used then this would be correct.

D (incorrect) If the time RHR-V-17B becomes inoperable is used and adding in the 24 hour extension then this would be correct.

Technical Reference(s): Tech Spec 3.6.1.5, Tech Spec 1.3 Completion Times
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: TS 3.6.1.5

Learning Objective: 5783 (As available)

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 H / 3

10 CFR Part 55 Content: 55.41 _____
55.43 2

Comments:

Key change for this question

Examination Outline Cross-Reference	Level	SRO
234000 Fuel Handling Equipment	Tier	2
	Group	2
K5. Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT	K/A #	234000.K5.02
	Rating	3.7
	Rev / Date	2
K5.02 †Fuel handling equipment interlocks		

Proposed Question: SRO – 18 (93)

Answer the following questions regarding refueling bridge hoist interlocks:

1) The _____ Interlock is required to be operable by Technical Specifications during in-vessel fuel movement.
2) According to the Technical Specification Bases document, this interlock is intended to _____.

A. 1) Fuel Hoist
2) ensure grapple is fully closed to prevent fuel damage when lifting a fuel bundle

B. 1) Main Hoist Fuel Loaded
2) ensure grapple is fully closed to prevent fuel damage when lifting a fuel bundle

C. 1) Fuel Hoist
2) prevent a prompt reactivity excursion during refueling could potentially result in fuel failure

D. 1) Main Hoist Fuel Loaded
2) prevent a prompt reactivity excursion during refueling could potentially result in fuel failure

Proposed Answer: C, D

Explanation (Optional):

A (incorrect) 1) First part is correct – the Fuel Hoist Interlock is required to be operable by technical specifications.

2) Second part is incorrect as the Fuel Hoist interlock is not related to the grapple function. Plausible if applicant confused Fuel Hoist Interlock with Main Hoist Fuel Loaded (Hoist Loaded) Interlock which is designed to prevent lifting a partially grappled fuel assembly which could result in fuel damage if the partially grappled fuel assembly is subsequently dropped.

B (incorrect) 1) First part is incorrect – the Main Hoist Fuel Loaded (Hoist Loaded) Interlock is not required by technical specifications.

2) Second part is incorrect but plausible since the Main Hoist Fuel Loaded (Hoist Loaded) interlock is designed to prevent lifting a partially grappled fuel assembly which could result in fuel damage if the partially grappled fuel assembly is subsequently dropped.

C (CORRECT) 1) First part is correct –The Fuel Hoist Interlock (which is given the term "Refueling platform fuel grapple fuel loaded" interlock) is required by technical specifications on the basis that it prevents in-vessel fuel loading with any control rod not fully inserted. This term is used in System Description SD000207 page 27 of 51. Additionally, the term "Main hoist" is also used to describe the same interlock. The term Fuel Hoist is not used in Technical Specification SR 3.9.1.1 (page 3.9.1.2) or Technical Specification Bases (page B 3.9.1-1) to describe this interlock.

2) This interlock is intended to prevent a prompt reactivity excursion during fuel loading. SRO-only requirement met due to fuel handling and tech spec bases aspects to the question.

D (CORRECT) 1) First part is correct – The term "Main Hoist Fuel Loaded" more closely describes the terms used in SR 3.9.1.1 "fuel loaded" and the technical specification bases B 3.9.1 "fuel grapple (main hoist)". Therefore this term should be accepted to describe the technical specification required surveillance in SR 3.9.1.1 c. The term "Main hoist fuel loaded" will be modified to "Hoist Loaded" in SD000207 page 27 of 51 to remove any future confusion and provide alignment with the term used in the LCS surveillance OSP-NSSE-C401 which is used to test the SD000207 described interlock as required per LCS SR 1.9.1.6.

2) Second part is correct (see answer C above).

Technical Reference(s): TS 3.9.1, TS 3.9.1 Bases, OSP-NSSE-C401, Rev 8, page 9-10, SD000207, Fuel Handling, Rev 12, page 27.
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5362 (As available)

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 F / 3

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 _____

55.43 2

Comments: Submitted for consideration of two correct answers: C or D. Justification is provided above. NRC panel agrees and accepted both answers as correct since they are not directly opposed and justification of terminology differences is acceptable.

Examination Outline Cross-Reference	Level	SRO
2.1 Conduct of Operations	**Tier**	3
	Group	N/A
2.1.29 Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	**K/A #**	2.1.29
	Rating	4.0
	Rev / Date	2

Proposed Question: SRO – 19 (94)

Maintenance on a system has just completed that affected a valve on the Locked Valve Checklist, Attachment 7.1 of PPM 1.3.29. The valve was **NOT** verified in its locked position following maintenance, however the valve can be verified closed by plant parameters. The valve is located in a Radiation Area where the exposure to one individual checking the position of the valve will be approximately 12 mrem.

Which of the following is required to verify the position of the valve?

A. Perform a partial valve lineup using the Locked Valve Checklist; the independent verification must be performed.

B. Perform a partial valve lineup using the Locked Valve Checklist and the independent verification may be waived by the Shift Manager.

C. Perform a partial valve lineup using the Locked Valve Checklist. The CRS/Shift Manager completes Attachment 7.2 (Deviation from Locked Valve Checklist) to document waiving the independent verification.

D. Performance of a valve lineup can be waived by the Shift Manager until after the next outage by completing Attachment 7.2 (Deviation From Locked Valve Checklist).

Proposed Answer: B

Explanation (Optional):
Per PPM 1.3.29 Page 4 "When significant radiation exposures are likely to occur as a result of performing an independent verification, the independent verification can be waived by the Shift Manager. As a guideline, an estimated exposure of 5 mrem for a single (one valve) verification task is considered a significant radiation exposure. However, in these situations an alternate means for independent verification that does not involve radiation exposure (e.g., observing process parameters) should be considered."

A (incorrect) The independent verification may be waived.

B (CORRECT) The waiver of the independent verification by the Shift manager is permitted due to expected dose >5 mrem.

C (incorrect) Attachment 7.2 is used to document repositioning of a component, not waiver of lineup requirements. 1.3.29 section 5.3 does not allow waiver of the positioning of the valve and installation of the locking device.

D (incorrect) The Independent verification is not a deviation from the intended valve position. The Attachment 7.2 is not used for this case.

Technical Reference(s): PPM 1.3.29 Rev 069
(Attach if not previously provided, including version/revision number) _____

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3038 (As available)

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 F / 3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____ 55.45.1
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	SRO
2.1 Conduct of Operations	**Tier**	3
	Group	N/A
G2.1.35 Knowledge of the fuel-handling responsibilities of SROs.	**K/A #**	2.1.35
	Rating	3.9
	Rev / Date	2

Proposed Question: SRO – 20 (95)

When in Mode 5, which of the following evolutions **MUST** have a Core Alteration Supervisor present?

A. Raising an irradiated fuel bundle in the fuel prep machine.

B. Movement of a new fuel bundle in the spent fuel pool.

C. Movement of individual fuel pins for in-pool inspection of irradiated fuel assemblies.

D. Movement of an irradiated fuel bundle in the spent fuel pool.

Proposed Answer: D

Explanation (Optional):
Per PPM 6.3.2 "The Core Alt Supervisor shall hold a Senior Operators License and shall have no other concurrent responsibilities during refueling."

Per SWP-RXE-01 "The Core Alt Supervisor (RFAE) shall directly supervise all movement of irradiated fuel during Mode 5. Fuel movements in other Modes shall be directly supervised by either a Core Alt Supervisor (RFAE) or a Spent Fuel Pool Supervisor (RFAF) (except not in Mode 5). The only exceptions to this requirement are 1) the movement of individual fuel pins or subassemblies associated with in-pool inspection of irradiated fuel assemblies, and 2) the raising or lowering of irradiated fuel bundles in the fuel prep machines. Supervision of these activities by either a Core Alt Supervisor (RFAE) or a Spent Fuel Pool Supervisor (RFAF) is not required.

A (incorrect) See SWP-RXE-01 Statement above.

B (incorrect) See SWP-RXE-01 Statement above.

C (incorrect) See SWP-RXE-01 Statement above.

D (CORRECT) See SWP-RXE-01 Statement above.

Technical Reference(s): SWP-RXE-01 pg. 21, PPM 6.3.2 Rev. 23 pg. 13, 28
(Attach if not previously provided,

including version/revision number) _____

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3038 (As available)

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 F / 4

10 CFR Part 55 Content: 55.41 _____

55.43 7

Comments:

Examination Outline Cross-Reference	Level	SRO
2.2 Equipment Control	**Tier**	3
	Group	N/A
2.2.5 Knowledge of the process for making design or operating changes to the facility.	**K/A #**	2.2.5
	Rating	3.2
	Rev / Date	2

Proposed Question: SRO-21 (96)

The RCIC keep fill pump will **NOT** automatically start due to a failed pressure switch. To maintain RCIC operability, the control switch for RCIC-P-3 will be placed in RUN instead of AUTO. A new pressure switch will be installed and calibrated in approximately two weeks. Prior to implementation, a 10 CFR 50.59 review _____ and a _____ is prepared to track the configuration change.

A. Is **NOT** required, Component Status Control Order

B. Is **NOT** required, Caution Tag

C. IS required, Component Status Control Order

D. IS required, Caution Tag

Proposed Answer: D

Explanation (Optional):

A (incorrect) A 50.59 screening is required because this is an activity required to supplement an SSC design function. A caution tag must be used but it is plausible that a the Component Status Control Order (CSCO) tag could be used because the CSCO process is used for other temporary configuration changes, including some related to degraded conditions. That a 50.59 screening might not be required is plausible due to the RUN position being an available mode for the RCIC keep-fill pump.

B (incorrect) A 50.59 screening is required, plausible due to the RUN position being an available mode for the RCIC keep-fill pump. Also plausible because PPM 1.3.9 does allow caution tags to be used to control component repositioning.

C (incorrect) A 50.59 screening is required, but a CSCO tag cannot be used. The plausibility of this is described in A.

D (CORRECT) A 50.59 screening is required because this is an activity required to supplement an SSC design function. A caution tag must be used per PPM 1.3.9.

Technical Reference(s): PPM 1.3.9 Rev 51 page 21, SWP-CM-02 Rev 0 page 11
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7845 (As available)

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 H / 2
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 10
55.43 _____

Comments:

Examination Outline Cross-Reference	Level	SRO
2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	**Tier**	3
	Group	N/A
	K/A #	2.2.25
	Rating	4.2
	Rev / Date	2

Proposed Question: SRO-22 (97)

Concerning the Reactor Coolant System Pressure Safety Limit:

Reactor steam dome pressure is required to be ≤ _____ psig, and is limited by the _____.

A. 1325; Reactor Recirc Suction Piping

B. 1325; Reactor Pressure Vessel

C. 1375; Reactor Recirc Suction Piping

D. 1375; Reactor Pressure Vessel

Proposed Answer: B

Explanation (Optional):

A (incorrect) The first part of the distractor is correct. Per TS Safety Limit 2.1.2, Reactor steam dome pressure is required to be ≤ 1325 psig. The Tech Spec bases states that the RPV is designed to an ASME code which permits a maximum pressure transient of 110% (1375 psig) of design pressure (1250 psig). The safety limit value of 1325 psig as measured at the steam dome is equivalent to 1375 psig at the lowest elevation of the RPV.

The second part of the distractor is incorrect. The Reactor Recirc suction piping is plausible, because its design pressure is also 1250 psig, but ASME code permits 125% of design pressure (1562 psig) for this piping. The pressure safety limit is selected to be the lowest transient overpressure allowed by the applicable code (i.e. RPV).

B (CORRECT) Per TS Safety Limit 2.1.2, Reactor steam dome pressure is required to be ≤ 1325 psig. The Tech Spec bases states that the RPV is designed to an ASME code which permits a maximum pressure transient of 110% (1375 psig) of design pressure (1250 psig). The safety limit value of 1325 psig as measured at the steam dome is equivalent to 1375 psig at the lowest elevation of the RPV. The Reactor Recirc suction piping design pressure is also 1250 psig, but ASME code permits 125% of design pressure (1562 psig) for this piping. The pressure safety

limit is selected to be the lowest transient overpressure allowed by the applicable code.

C (incorrect) The first part of the distractor is incorrect. Per TS Safety Limit 2.1.2, Reactor steam dome pressure is required to be ≤ 1325 psig. The Tech Spec bases states that the RPV is designed to an ASME code which permits a maximum pressure transient of 110% (1375 psig) of design pressure (1250 psig). The safety limit value of 1325 psig as measured at the steam dome is equivalent to 1375 psig at the lowest elevation of the RPV.

1375 psig is plausible because it is the pressure upon which the safety limit is based. However, because RPV pressure is monitored in the steam dome, and not at the bottom head, the safety limit is 1325 psig.

D (incorrect) The first part of the distractor is incorrect. Per TS Safety Limit 2.1.2, Reactor steam dome pressure is required to be ≤ 1325 psig. The Tech Spec bases states that the RPV is designed to an ASME code which permits a maximum pressure transient of 110% (1375 psig) of design pressure (1250 psig). The safety limit value of 1325 psig as measured at the steam dome is equivalent to 1375 psig at the lowest elevation of the RPV.

The second part of the distractor is incorrect. The Reactor Recirc suction piping is plausible, because its design pressure is also 1250 psig, but ASME code permits 125% of design pressure (1562 psig) for this piping. The pressure safety limit is selected to be the lowest transient overpressure allowed by the applicable code (i.e. RPV).

Technical Reference(s): TS 2.1.2; TS Bases 2.1.2-1 and -2
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: 6925 (As available)

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 F3
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 2

Comments:

Examination Outline Cross-Reference	Level	SRO
2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.	Tier	3
	Group	N/A
	K/A #	2.3.4
	Rating	3.7
	Rev / Date	2

Proposed Question: SRO-23 (98)

A severe plant transient is in progress. Conditions require a task to be performed in the Reactor Building that will protect some very expensive equipment, and an individual has volunteered to perform the task. Performance of the task will cause the performer to exceed the 10 CFR Part 20 limit for TEDE.

At what emergency classification are the Energy Northwest administrative exposure hold points automatically waived and, as the Emergency Director, what is the **MAXIMUM** dose you are authorized to allow the individual to receive in this situation?

A. Alert; 10 rem TEDE

B. Alert; 25 rem TEDE

C. Site Area Emergency; 10 rem TEDE

D. Site Area Emergency; 25 rem TEDE

Proposed Answer: A

Explanation (Optional):

A (CORRECT) Per PPM 13.2.1, "Declaration of an Alert or higher emergency classification automatically waives Energy Northwest administrative exposure hold points." The stem states that the task being performed which will result in exceeding the 10 CFR Part 20 limit for TEDE is being performed to protect very expensive equipment. The EPA 400 PAGs for emergency workers establish a dose limit of 10 rem to protect valuable property.

B (incorrect) Per PPM 13.2.1, "Declaration of an Alert or higher emergency classification automatically waives Energy Northwest administrative exposure hold points." The stem states that the task being performed which will result in exceeding the 10 CFR Part 20 limit for TEDE is being performed to protect very expensive equipment. The EPA 400 PAGs for emergency workers establish a dose limit of 10 rem to protect valuable property, which makes the second part of this distractor incorrect. 25 rem is plausible because it is the dose limit established in the PAGs for life-saving or protection of large populations.

C (incorrect) The first part of the distractor is incorrect, because it is the Alert level, not the SAE level, at which the administrative hold point is waived. SAE is plausible because there are automatic actions (PARS) that occur at this emergency level. The second part of the distractor is correct. The stem states that the task being performed which will result in exceeding the 10 CFR Part 20 limit for TEDE is being performed to protect very expensive equipment. The EPA 400 PAGs for emergency workers establish a dose limit of 10 rem to protect valuable property.

D (incorrect) The first part of the distractor is incorrect, because it is the Alert level, not the SAE level, at which the administrative hold point is waived. SAE is plausible because there are automatic actions (PARS) that occur at this emergency level. The stem states that the task being performed which will result in exceeding the 10 CFR Part 20 limit for TEDE is being performed to protect very expensive equipment. The EPA 400 PAGs for emergency workers establish a dose limit of 10 rem to protect valuable property, which makes the second part of this distractor incorrect. 25 rem is plausible because it is the dose limit established in the PAGs for life-saving or protection of large populations.

Technical Reference(s): PPM 13.2.1 Rev.022, pg. 5 and 11
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: 6019, 6020 (As available)

Question Source: Bank # SRO-23
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam October 2009
(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 F2
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
55.43 4

Comments:

Examination Outline Cross-Reference	Level	SRO
2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	**Tier**	3
	Group	N/A
	K/A #	2.3.12
	Rating	3.7
	Rev / Date	2

Proposed Question: SRO-24 (99)

With one rod withdrawn in Mode 5, SR 3.9.2.1 requires the mode switch to be verified in the locked position every 12 hours in order to meet the requirements of TS 3.9.2 Refuel Position One-Rod-Out Interlock.

1) How does the Control Room ensure this requirement is met?
2) What does this interlock protect against?

A. 1) By verifying OSP-INST-H103 (Shift and Daily Instrument Checks (Mode 5) is current.
2) A prompt reactivity excursion during refueling which could potentially result in fuel failure with subsequent release of radioactive material to the environment.

B. 1) By verifying OSP-NSSE-W402 (Refuel Position One-Rod-Out Interlock CFT) is current.
2) A prompt reactivity excursion during refueling which could potentially result in fuel failure with subsequent release of radioactive material to the environment.

C. 1) By verifying OSP-INST-H103 (Shift and Daily Instrument Checks (Mode 5)) is current.
2) A prompt criticality event which could result in high radiation exposure to operators on the refueling floor.

D. 1) By verifying OSP-NSSE-W402 (Refuel Position One-Rod-Out Interlock CFT) is current.
2) A prompt criticality event which could result in high radiation exposure to operators on the refueling floor.

Proposed Answer: A

Explanation (Optional):

A (CORRECT) 1) Correct - OSP-INST-H103 is the surveillance which verifies the mode switch is locked in the Refuel position every 12 hours.

2) Correct - Per TS bases 3.9.2 “A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment.”

B (incorrect) 1) Incorrect - OSP-NSSE-W402 is the surveillance which covers the Channel Functional Test for TS 3.9.2. and does include steps which verify the mode switch is in the Refuel position but is not performed every 12 hours.

2) Correct - Per TS bases 3.9.2 "A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment."

C (incorrect) 1) Correct - OSP-INST-H103 is the surveillance which verifies the mode switch is locked in the Refuel position every 12 hours.

2) Incorrect - A prompt criticality event could result in radiation exposure to personnel on the refuel floor, however the TS bases calls out the FSAR analysis consequences are release to the environment.

D (incorrect) 1) Incorrect - OSP-NSSE-W402 is the surveillance which covers the Channel Functional Test for TS 3.9.2. and does include steps which verify the mode switch is in the Refuel position but is not performed every 12 hours.

2) Incorrect - A prompt criticality event could result in radiation exposure to personnel on the refuel floor, however the TS bases calls out the FSAR analysis consequences are release to the environment.

Technical Reference(s): OSP-INST-H103 Rev. 039 page 8;
(Attach if not previously provided, including version/revision number) TS 3.9.2 and TS bases 3.9.2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 13287 (As available)

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 H / 4

10 CFR Part 55 Content: 55.41 _____
55.43 6

Comments:

Examination Outline Cross-Reference	Level	SRO
2.4 Emergency Procedures / Plan	**Tier**	3
	Group	N/A
2.4.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.	**K/A #**	2.4.30
	Rating	4.1
	Rev / Date	2

Proposed Question: SRO – 25 (100)

Concerning the systems listed in 50.72(b)(3)(iv)(B), which of the following lists events that are **BOTH** reportable to the NRC per PPM 1.10.1, Notifications and Reportable Events?

An actuation that occurs as the result of…

A. an intentional manual initiation based on plant conditions;
an intentional manual initiation as part of a pre-planned sequence during testing.

B. an intentional manual initiation as part of a pre-planned sequence during testing;
actual plant conditions with no testing in progress.

C. actual plant conditions that were **NOT** pre-planned during testing;
a test signal generated during a calibration check.

D. actual plant conditions with no testing in progress;
actual plant conditions that were **NOT** pre-planned during testing.

Proposed Answer: D

Explanation (Optional):

A (incorrect) An intentional manual initiation as part of a pre-planned sequence during testing is not reportable per 50.72.

B (incorrect) An intentional manual initiation as part of a pre-planned sequence during testing is not reportable per 50.72.

C (incorrect) A test signal generated during a calibration check is an invalid signal and is not reportable per 50.72

D (CORRECT) Actual plant conditions not pre-planned are reportable per 50.72.

Technical Reference(s): PPM 1.10.1 Pages 7 and 11; NUREG-1022 Rev.3 Pages 19-26
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6011 (As available)

Question Source: Bank # SRO-11
Modified Bank # _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2011
(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 H / 3

10 CFR Part 55 Content: 55.41 _____
55.43 5

Comments:

References for SRO only portion of Written Exam

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

----------NOTE----------
LCO 3.0.4.b is not applicable to HPCS.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days[1]
B High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC System is required to be OPERABLE. AND B.2 Restore HPCS System to OPERABLE status.	Immediately 14 days

[1] The Completion Time that one train of RHR (RHR-B) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-B from the modification activity. Upon successful restoration of RHR-B, this footnote is no longer applicable and will expire at 05:00 PST on February 9, 2015.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two ECCS injection subsystems inoperable. OR One ECCS injection and one ECCS spray subsystem inoperable.	C.1 Restore ECCS injection/spray subsystem to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours
E. One required ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days
F. One required ADS valve inoperable. AND One low pressure ECCS injection/spray subsystem inoperable.	F.1 Restore ADS valve to OPERABLE status. OR F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
G. Required Action and associated Completion Time of Condition E or F not met. OR Two or more required ADS valves inoperable.	G.1 Be in MODE 3. AND G.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. HPCS and Low Pressure Core Spray (LPCS) Systems inoperable. OR Three or more ECCS injection/spray subsystems inoperable. OR HPCS System and one or more required ADS valves inoperable. OR Two or more ECCS injection/spray subsystems and one or more required ADS valves inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.2	----------NOTE---------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than 48 psig in MODE 3, if capable of being manually realigned and not otherwise inoperable. ---------- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.1.3	Verify ADS accumulator backup compressed gas system average pressure in the required bottles is ≥ 2200 psig.	31 days
SR 3.5.1.4	Verify each ECCS pump develops the specified flow rate with the specified differential pressure between reactor and suction source. SYSTEM / FLOW RATE / DIFFERENTIAL PRESSURE BETWEEN REACTOR AND SUCTION SOURCE LPCS / ≥ 6200 gpm / ≥ 128 psid LPCI / ≥ 7200 gpm / ≥ 26 psid HPCS / ≥ 6350 gpm / ≥ 200 psid	In accordance with the Inservice Testing Program
SR 3.5.1.5	----------NOTE---------- Vessel injection/spray may be excluded. ---------- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	----------NOTE---------- Valve actuation may be excluded. ---------- Verify the ADS actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.7	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify each required ADS valve opens when manually actuated.	24 months on a STAGGERED TEST BASIS for each valve solenoid
SR 3.5.1.8	----------NOTE---------- ECCS actuation instrumentation is excluded. ---------- Verify the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is within limits.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.1.5 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.1.5 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1	Restore RHR drywell spray subsystem to OPERABLE status.	7 days[1]
B. Two RHR drywell spray subsystems inoperable.	B.1	Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours

[1] The Completion Time that one train of RHR (RHR-B) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-B from the modification activity. Upon successful restoration of RHR-B, this footnote is no longer applicable and will expire at 05:00 PST on February 9, 2015.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.5.1	Verify each RHR drywell spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.1.5.2	Verify each spray nozzle is unobstructed.	10 years

3.6 CONTAINMENT SYSTEMS

3.6.2.1 Suppression Pool Average Temperature

LCO 3.6.2.1 Suppression pool average temperature shall be:

a. ≤ 90°F when THERMAL POWER is > 1% RTP and no testing that adds heat to the suppression pool is being performed;

b. ≤ 105°F when THERMAL POWER is > 1% RTP and testing that adds heat to the suppression pool is being performed; and

c. ≤ 110°F when THERMAL POWER is ≤ 1% RTP.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool average temperature > 90°F but ≤ 110°F. AND THERMAL POWER > 1% RTP. AND Not performing testing that adds heat to the suppression pool.	A.1 Verify suppression pool average temperature ≤ 110°F. AND A.2 Restore suppression pool average temperature to ≤ 90°F.	Once per hour 24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to ≤ 1% RTP.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Suppression pool average temperature > 105°F. AND THERMAL POWER > 1% RTP. AND Performing testing that adds heat to the suppression pool.	C.1	Suspend all testing that adds heat to the suppression pool.	Immediately
D. Suppression pool average temperature > 110°F but ≤ 120°F.	D.1	Place the reactor mode switch in the shutdown position.	Immediately
	AND		
	D.2	Verify suppression pool average temperature ≤ 120°F.	Once per 30 minutes
	AND		
	D.3	Be in MODE 4.	36 hours
E. Suppression pool average temperature > 120°F.	E.1	Depressurize the reactor vessel to < 200 psig.	12 hours
	AND		
	E.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1.1	Verify suppression pool average temperature is within the applicable limits.	24 hours AND 5 minutes when performing testing that adds heat to the suppression pool

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electric Power Distribution System; and

b. Three diesel generators (DGs).

APPLICABILITY: MODES 1, 2, and 3.

----------NOTE----------
Division 3 AC electrical power sources are not required to be OPERABLE when High Pressure Core Spray System is inoperable.

ACTIONS

----------NOTE----------
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit. AND	1 hour AND Once per 8 hours thereafter

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant required feature(s)
	AND		
	A.3	Restore offsite circuit to OPERABLE status.	72 hours AND 6 days from discovery of failure to meet LCO when not associated with Required Action B.4.2.2 AND 17 days from discovery of failure to meet LCO
B. One required DG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.1 Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours if not performed within the past 24 hours
	AND	
	B.4.1 Restore required DG to OPERABLE status.	72 hours from discovery of an inoperable DG AND 6 days from discovery of failure to meet LCO
	OR	
	B.4.2.1 Establish risk management actions for the alternate AC sources.	72 hours
	AND	
	B.4.2.2 Restore required DG to OPERABLE status.	14 days AND 17 days from discovery of failure to meet LCO

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND	
	C.2 Restore one offsite circuit to OPERABLE status.	24 hours
D. One offsite circuit inoperable. AND One required DG inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any division. ----------	
	D.1 Restore offsite circuit to OPERABLE status.	12 hours
	OR	
	D.2 Restore required DG to OPERABLE status.	12 hours
E. Two required DGs inoperable.	E.1 Restore one required DG to OPERABLE status.	2 hours OR 24 hours if Division 3 DG is inoperable

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	12 hours 36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

1.7 PLANT SYSTEMS

1.7.1 Area Temperature Monitoring

RFO 1.7.1 Area temperatures shall be maintained within limits as shown in Table 1.7.1-1.

APPLICABILITY: When equipment in a room or area listed in Table 1.7.1-1 is required to be OPERABLE.

COMPENSATORY MEASURES

----------NOTE----------
Separate condition entry is allowed for each area.

CONDITION	REQUIRED COMPENSATORY MEASURE		COMPLETION TIME
A. ----------NOTE---------- Required Compensatory Measure A.2 shall be completed if this Condition is entered. ---------- With one or more areas not within limits of Table 1.7.1-1.	A.1	Enter the condition referenced in Table 1.7.1-1.	Immediately
	AND		
	A.2	Initiate a Condition Report (CR).	24 hours
B. As required by Compensatory Measure A.1 and referenced in Table 1.7.1-1.	B.1	Initiate action to restore area or room temperature to be within the Condition B limits of Table 1.7.1-1.	Immediately
	AND		
	B.2	Perform SR 1.7.1.1 for affected areas.	Once per 4 hours
C. As required by Compensatory Measure A.1 and referenced in Table 1.7.1-1.	C.1	Restore area or room temperature to be within the limits of Table 1.7.1-1.	1 hour