Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A		215004 K1.02 Source Range Monitor System				Importance		3.4	
Statement		Knowledge of the physical connections and/or cause-effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and Reactor manual control.							

QUESTION 1

Unit 1 startup is in progress.

The reactor is critical with a 300-second period.

While operators are withdrawing SRMs per GO-200-002, annunciator ROD OUT BLOCK (AR-104-H03) is received.

Operators stop withdrawing SRMs and note the following SRM readings:

SRM	Counts (cps)	Position
A	90	Partially withdrawn
B	200	Partially withdrawn
C	8E4	Fully inserted
D	2E5	Fully inserted

IRMs are reading 10 on Range 2.

Which one of the following identifies the actions that will clear the ROD OUT BLOCK alarm and allow control rod withdrawal to continue?

A. Bypass SRM D, ONLY

B. Insert SRM A to obtain approximately 1000 cps, ONLY

C. Place all IRMs on Range 3
Bypass SRM D

D. Insert SRM A to obtain approximately 1000 cps
Bypass SRM D

Proposed Answer D

Applicant References None

Explanation SRMs A and D are generating rod-out block signals to the RMCS. SRM A is reading below the WITHDRAW PERMIT setpoint of 100 cps and is not fully inserted. SRM D is reading above the UPSCALE setpoint of 1E5 cps. A rod-out block from ANY SRM channel to RMCS generates a RMCS ROD OUT BLOCK to prevent control rod withdrawal.

D is the correct answer. Inserting SRM A will clear the WITHDRAW PERMIT rod-block signal from SRM A to RMCS, and bypassing SRM D will clear the SRM UPSCALE rod-block signal from SRM D.

A is incorrect. Bypassing SRM D will clear its rod-out block signal to RMCS, but the signal from SRM A remains.

B is incorrect. Inserting SRM A to obtain 1000 cps, per the applicable GO-200-002 guidance, will clear its rod-out block signal to RMCS, but the signal from SRM D remains.

C is incorrect. While placing all IRMs on Range 3 will bypass the WITHDRAW PERMIT rod-out block from SRM A, it will result in a DOWNSCALE trip from all IRMs and a rod-out block signal to RMCS. Bypassing SRM D would clear the rod-out block signal to RMCS, but to no effect.

10CFR55	41.6
Technical References	AR-104-E06 AR-104-B06 AR-104-C05 GO-100-002
Learning Objectives	1345
Question Source	New
Previous NRC Exam	No
Comments	K/A sampled on LOC25 NRC exam. This question satisfies the significantly modified critieria of NUREG-1021 ES-401 D.2.f

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SUSQUEHANNA STEAM ELECTRIC STATION
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Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		205000 K1.05 Shutdown Cooling System (RHR Shutdown Cooling Mode)				Importance		3.9	
Statement		Knowledge of the physical connections and/or cause-effect relationships between SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) and the following: LPCI.							

QUESTION 2

Unit 1 is in Mode 3 performing a unit shutdown for a failing Recirc Pump seal.

RHR Loop A has just been placed into Shutdown Cooling using RHR Pump 1A.

The recirc pump seal fails completely. Drywell pressure rises and a RPS trip on high Drywell pressure occurs.

Which one of the following describes the response of RHR?

	RHR Loop A	RHR Loop B
A.	RHR Pump 1A tripped RHR SDC isolated	Injecting in LPCI alignment
B.	RHR Pump 1A running in SDC RHR Pump 1C in standby	Standby
C.	RHR Pumps 1A and 1C tripped RHR SDC isolated	Running on minimum flow
D.	RHR Pumps 1A and 1C running in SDC	Injecting in LPCI alignment

Proposed Answer **D**

Applicant References **None**

Explanation **A high Drywell pressure LOCA initiation signal has been received. With reactor pressure below the SDC interlock of 98 psig this results in a LPCI initiation signal to both divisions of RHR. RHR Loop B will start, align for LPCI, and inject to the reactor with reactor pressure well below the 430 psig injection valve auto-open permissive. The SDC flowpath is unaffected by Drywell pressure, the only effects on RHR Loop A is that RHR Pump 1C will start in the SDC alignment in addition to RHR Pump 1A and HV-151-F017A (LPCI o/b inj valve) will receive a full-open signal.**

A Incorrect. While RHR Loop B will inject in the LPCI alignment, RHR Pump 1A will not receive a trip signal as a SDC isolation does not occur on high DW pressure.

B Incorrect. RHR Loop B will align for and inject in the LPCI mode. RHR Pump 1C will start in the SDC lineup, as the F006C is opened as part of the procedure for placing RHR Loop A in service, regardless of the RHR pump started.

C Incorrect. RHR Loop B will align for and inject in the LPCI mode. There is no SDC isolation signal, so neither RHR Loop A pump receives a trip signal.

D **Correct. RHR Pump 1C will start on the high DW pressure /low reactor pressure combination. The RHR Loop A SDC lineup is unaffected by the DW pressure signal. RHR Loop B will align for and inject in the LPCI mode.**

10CFR55	**41.7**	
Technical References	**OP-149-002 Step 2.1.2.g-l, 2.1.7, 2.6.3.a NOTE**	
Learning Objectives	**10766 u**	
Question Source	**Bank**	**ILO LXR TMOP049/180/002**
Previous NRC Exam	**No**	
Comments		

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Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A	212000 K2.02 Reactor Protection System					Importance	2.7		
Statement	Knowledge of electrical power supplies to the following: Analog trip system logic cabinets								

QUESTION 3

Which one of the following identifies the power supply(s) that if de-energized, would result in venting the scram air header through the Backup Scram Valve(s) SV-147F110A(B)?

A. 1Y201A AND 1Y201B

B. 1Y201A AND 1D614

C. 1D614 AND 1D624

D. 1D614, ONLY

Proposed Answer	**A**
Applicant References	**None**
Explanation	**The Backup Scram Valves SV-147110A(B) are energize-to-open, DC-powered solenoid valves that individually provide a redundant means to vent the scram air header on actuation. The valve solenoids are energized by the respective DC power supply 1D614(624) on a full RPS initiation.** **A is the correct answer. De-energization of the RPS Buses 1Y201A and 1Y201B removes power from the RPS relay logic cabinets 1C609 and 1C611 and deenergizes the RPS K14x trip relays resulting in a full RPS initiation signal which energizes the Backup Scram Valve solenoids.** **B, C and D are all incorrect as loss of DC power to the Backup Scram Valves prevent the valve from actuating.** **B is plausible as this is the power supplies to the RPS A trip system and Backup Scram Valve SV-147110A.** **C is plausible as this choice represents the loss of power to both divisions of Backup Scram valves.** **D is plausible as this choice is the power supply to Backup Scram Valve SV-147110A and represents incorrect application of the deenergize-to-open operating principle of the scram pilot solenoid valves to the Backup Scram Valves.**
10CFR55	**41.7**
Technical References	**M-147 Sht 1** **M1-C72-22 Sht 1,12,17** **TM-OP-058**
Learning Objectives	**10072**
Question Source	**New**
Previous NRC Exam	**No**
Comments	**3/13 rat. Minor editorial corrections, swapped C&D distractors, based on Ops Reviewer comments.**

Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	4
K/A		259002 K3.07 Reactor Water Level Control System				Importance		3.4	
Statement		Knowledge of the effect that a loss or malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following: Reactor water level indication							

QUESTION 5

Use your provided references to answer this question.

Unit 1 is operating at rated power.

The following reactor level indications are observed on the 1C652 Standby Information Panel.

Narrow Range A	+31 in, down slow
Narrow Range B	+39 in, up slow
Narrow Range C	+31 in, down slow
Wide Range	+18 in, down slow
Narrow Range (XR-10602)	+35 in, steady
Upset Range (XR-10602)	+32 in, down slow

Wide Range indications on 1C601 also show +18 in, down slow.

Which one of the following identifies all correct level indication(s) in these conditions?

A. Narrow Range B

B. Upset Range (XR-10602)
Wide Range

C. Narrow Range A and C
Wide Range

D. Narrow Range A and C
Upset Range (XR-10602)
Wide Range

Proposed Answer: D

Applicant References: ON-145-001 Att A

Explanation: The indications provided are consistent with a slow failure high of the Narrow Range B (NRLBB) signal in the Feedwater Level Control System. The signal has not yet drifted high enough for it to be flagged as DEVIANT, so the FWLC Average Level input is still taken from NRLA and NRLBB and the low median level. FWLC Selected Level remains Average Level. The XR-10601 NR indication is the FWLC Selected Level. Because of the simple arithmetic average as NRLBB drifts up FW flow is reduced to return Average Level to the FWLC setpoint of +35", resulting in all valid reactor level indicators slowly indicating lower as FW flow to the reactor is reduced.

A Incorrect. Narrow Range B has drifted high. The other level indications provided are associated with both the C004 and C005 instrument racks, eliminating a common-mode failure due to variable/reference leg leaks or condensing chamber issues.

B Incorrect. While UR and WR are correct, NR A and C are also correct.

C Incorrect. While NR A and C, and WR, are all correct, UR is also correct.

D CORRECT. NR A and C, UR, and WR indications are all correct for the given conditions.

10CFR55	41.7
Technical References	ON-145-001 Section 2.0
Learning Objectives	15999
Question Source	New
Previous NRC Exam	No
Comments	

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Exam	**RO**	Tier	**2**	Group	**1**	Cognitive Level	**High**	Level of Difficulty	**3**
K/A	**211000 K3.01 Standby Liquid Control System**					Importance	**4.3**		
Statement	**Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following: Ability to shutdown the reactor in certain conditions**								

QUESTION 6

Unit 1 experienced a high-power ATWS.

Standby Liquid Control Pump A was started.

A local operator reports that the pump discharge relief valve lifted and is stuck open.

Which one of the following describes the availability of SLC to inject boron to shutdown the reactor under these conditions?

A. Boron is being injected to the reactor at the normal flowrate

B. Boron is being injected to the reactor at a reduced flowrate

C. SLC Pump B must be started to inject boron to the reactor

D. Boron can be injected to the reactor with RCIC, ONLY

Proposed Answer: **C**

Applicant References: **None**

Explanation: **Each SLC pump is provided with a discharge pressure relief valve located between the pump and its discharge check valve. The relief valve returns to the pump suction and is capable of passing full flow from the pump.**

- A **Incorrect. All flow from SLC Pump A is being returned to the pump suction via the lifted relief valve.**
- B **Incorrect. All flow from SLC Pump A is being returned to the pump suction via the lifted relief valve.**
- C **Correct. The SLC Pump A discharge check valve will seat to prevent flow from SLC Pump B passing through the open SLC Pump A relief valve. Starting SLC Pump B fires the 2nd squib valve creating a second flow path out of the SLC system to the reactor.**
- D **Incorrect. The SLC Pump A discharge check valve will seat to prevent flow from SLC Pump B passing through the open SLC Pump A relief valve. The B squib valve can be fired to create a second flow path. Use of RCIC is not required.**

10CFR55: **41.6**

Technical References: **M-148**

Learning Objectives: **10887 j**

Question Source: **Bank** **ILO LXR TMOP053/1214/006**

Previous NRC Exam: **No**

Comments

Exam	RO	Tier	2	Group	1	Cognitive Level	Low	Level of Difficulty	
K/A		239002 K2.01 Safety Relief Valves				Importance		2.8	
Statement		Knowledge of electrical power supplies to the following: SRV solenoids							

QUESTION 4

Which one of the following identifies all of the power-operated SRV functions that remain available on a loss of 1D614?

A. ADS initiation
Lower Relay Room manual operation

B. ADS initiation
Control Room manual operation

C. Control Room manual operation
Remote Shutdown Panel manual operation

D. Lower Relay Room manual operation
Remote Shutdown Panel manual operation

Proposed Answer: **A**

Applicant References: **None**

Explanation: **1D614 supplies power to the normal operation SRV solenoids and the Division 1 ADS logic and associated Division 1 ADS solenoids on the SRVs. The Remote Shutdown Panel handswitches also receive power from 1D614 to operate the normal operation SRV solenoids of the A, B and C SRVs. Division 2 of ADS is unaffected and upon an automatic or manual ADS initiation will energize the Division 2 ADS solenoids to open the ADS SRVs. The handswitches in the Lower Relay Room are part of the Division 2 ADS logic and will also function to open the SRVs via the Division 2 ADS logic and associated power supply.**

A **CORRECT. An ADS initiation and manual operation from the Lower Relay Room are still posible on a loss of 1D614. No other means of electrically operating the SRVs is available.**

B **Incorrect. Control Room manual operation is not possible as power is lost to the normal operating solenoids and the Control Room handswitches.**

C **Incorrect. Neither Control Room nor RSDP manual operation is possible as power is lost to the normal operating solenoids and both the Control Room and RSDP handswitches.**

D **Incorrect. While the ADS SRVs may be operated from the Lower Relay Room, RSDP manual operation is not possible as power is lost to the normal operating solenoids and the RSDP handswitches.**

10CFR55: **41.7**

Technical References: **E-180 Sht 1**
M1-B21-129 Sht 4, 5

Learning Objectives: **1651**

Question Source: **New**

Previous NRC Exam: **No**

Comments

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REACTOR OPERATOR WRITTEN EXAMINATION

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Operations Reviewer mJ / 03JUN14
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Exam	RO	Tier	2	Group	1	Cognitive Level	Low	Level of Difficulty	3
K/A		263000 K4.02 D.C. Electrical Distribution				Importance		3.1	
Statement		Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Breaker interlocks, permissives, bypasses and cross-ties							

QUESTION 7

The Class 1E 125V DC system automatically provides an alternate control power supply to select ESS Bus breakers to ensure LOOP/LOCA load shed occurs.

Which one of the following identifies loads that have the alternate power supply?

A. ESW Pump C
ESW Pump D
RHRSW Pump 1A
RHRSW Pump 1B

B. ESW Pump A
ESW Pump B
RHRSW Pump 2A
RHRSW Pump 2B

C. CRD Pump 1B
CRD Pump 2B
RHR Pump 1D
RHR Pump 2D

D. Core Spray Pump 1C
Core Spray Pump 1D
RHRSW Pump 1A
RHRSW Pump 1B

Proposed Answer **A**

Applicant References **None**

Explanation **The alternate breaker trip power supply logic is provided to ensure that specific loads are shed to prevent overloading a Diesel Generator when re-energizing its respective bus during a LOOP/LOCA with a failure of the 1D620 DC power supply. The alternate trip power is interlocked with the normal breaker control power to ensure the 2 DC sources are not cross-tied.**

A **Correct. These breakers required redundant trip capability.**

B **Incorrect. These are the equivalent 1A/1B and 2A/2B ESS Bus loads.**

C **Incorrect. While CRD Pumps 1B and 2B have the redundant trip power, no ECCS pumps do.**

D **Incorrect. While RHRSW Pumps 1A and 1B have the redundant trip power, no ECCS pumps do.**

10CFR55 **41.7**

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Technical References	**ON-102-610,620 TM-OP-002**	
Learning Objectives	**11859 e**	
Question Source	**Bank**	**ILO LXR TMOP002/10144/008**
Previous NRC Exam	**No**	
Comments		

Operations Reviewer mJ / 03JUN14
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Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A	**218000 K4.02 Automatic Depressurization System**					Importance		**4.0**	
Statement	**Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: Allows manual initiation of ADS logic**								

QUESTION 8

Unit 2 experienced a loss of all high-pressure reactor injection systems.

Both divisions of ADS were inhibited when the ADS logic timer alarms initiated without a valid initiation signal present.

Subsequently, a Rapid Depressurization on low reactor water level is required

Which one of the following identifies the action(s) required, if any, to <u>immediately</u> initiate ADS from the Control Room using the arm-and-depress pushbuttons?

A. ADS can be manually initiated immediately with no additional action

B. Start at least 1 RHR or 2 Core Spray pumps in a division

C. Un-inhibit ADS

D. Start at least 1 RHR or 2 Core Spray pumps in each division
<u>AND</u>
Un-inhibit ADS

Proposed Answer **A**

Applicant References **None**

Explanation **ADS has been inhibited due to an unspecified logic malfunction. Subsequently, reactor level has fallen below -161" requiring Rapid Depressurization. The ECCS initiation at -129" will start all low-pressure ECCS pumps and provide a valid initiation signal to ADS after a time delay.**

A **Correct. ADS can be manually initiated as long as 1 RHR or 2 Core Spray pumps in the associated division are running, which is the case as level has fallen below the -129" ECCS initiation setpoint. Depressing the manual initiation PB will result in immediate actuation of ADS and opening SRVs.**

B **Incorrect. The required pumps are already running due to reactor level < -129".**

C **Incorrect. This will initiate ADS, but after a 105-second time delay at minimum, and potentially signfiicantly longer if a high DW pressure signal is not present and the -129" reactor level low timer to bypasss the required DW pressure signal has only recently initiated.**

D **Incorrect. The required pumps are already running and the manual initiation PBs are not affected by the ADS inhibit keyswitch.**

10CFR55 **41.7**

Technical References **M1-B21-102 Sht 204**

Learning Objectives **2105 a,b**

Question Source **Bank** **LXR LOR TMOP083E/2105/005**

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Previous NRC Exam No

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Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A	215003 K5.01 Intermediate Range Monitor (IRM) System				Importance		2.6		
Statement	Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : Detector operation								

QUESTION 9

Unit 1 is starting up, with IRMs on Range 1 and 2.

Engineering just reported that the high-voltage power supplies on the Division 2 IRMs were mis-calibrated during the outage.

The Division 2 IRMs are operating with detector voltages set to the SRM voltage.

Which one of the following describes the operational implications for the Division 2 IRMs?

A. Will eventually fail upscale as reactor power is raised to enter Mode 1

B. Are reading higher than Division 1 IRMs

C. Are reading lower than Division 1 IRMs

D. No effect from detector voltage error, as IRMs are ionization detectors

Proposed Answer: B

Applicant References: None

Explanation: The Division 2 IRMs are operating at a higher voltage than normal, at the same voltage as a SRM. The SRMs operate in the proportional region of the gas-filled detector curve. The reading from IRMs operating at the higher detector voltage will be higher than those operating at the correct voltage.

A Incorrect. IRM detectors have lower-enriched uranium and a lower gas pressure. With one division of IRMs inoperable, LCO requirements for the IRM function are not satisfied and power ascension will be limited. It will take more than the 12 hours allowed by the RPS TS to raise reactor power sufficiently to make the SRMs fail upscale. IRM readings at Mode 1 are typically on Range 10, and well below the upscale alarm setpoint.

B Correct. The higher applied voltage on the Division 2 IRM detectors will result in significantly higher readings from these detectors, substantially higher than the Division 1 detectors.

C Incorrect. The higher applied voltage on the Division 2 IRM detectors will result in significantly higher readings from these detectors, substantially higher than the Division 1 detectors. This distractor represents a misconception about whether the IRMs or SRMs operate at the higher voltage required to place the detector in the proportional region of the gas-filled detector curve.

D Incorrect. The nominal IRM voltage of 100 VDC is such that the IRMs operate in the ionization region of the gas-filled detector curve. The 350 VDC applied to an IRM will result in the detector entering the proportional region of the GFDC.

10CFR55: 41.2

Technical References: TM-OP-078B

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Learning Objectives	**2337 c**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

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Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A	206000 K5.02 High Pressure Coolant Injection System					Importance		2.8	
Statement	Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM : Turbine shaft sealing: BWR-2,3,4								

QUESTION 10

Unit 2 scrammed from rated power due to a loss of offsite power.

Both trains of Standby Gas Treatment System fail to start and cannot be manually started.

Which one of the following identifies the operational implications of placing HPCI in pressure control for these conditions?

A. Becomes air-bound due to the buildup of non-condensible gases

B. Isolates on turbine exhaust diaphragm rupture

C. Isolates on high room temperature

D. HPCI room radiation levels rise

Proposed Answer: **D**

Applicant References: **None**

Explanation: **SGTS accepts the discharge of the HPCI barometric condenser vacuum pump. This pump functions on a HPCI initiation signal to draw a slight vacuum on the HPCI barometric condenser tank to aid in condensing steam drains. With no flowpath to SGTS a pressure relieving valve will direct the discharge of the pump back to the barometric condenser. Collection of steam drains will be affected, but HPCI operability is not affected.**

A Incorrect. Collection of non-condensible gases in the HPCI main steam supply will not result in the turbine or HPCI pump becoming air-bound.

B Incorrect. Additional moisture may be present in the HPCI turbine steam lines, which could carry into the turbine and exhaust. However, the steam drains will still function to remove moisture, albeit at a degraded efficiency. No concern exists for overpressurization of the HPCI turbine exhaust due to moisture.

C Incorrect. Steam leakage from the HPCI turbine seals will rise, but the HPCI isolation on high room temperature is sized for a 25 gpm steam leak.

D Correct. Steam leakage from the HPCI turbine seals will rise, resulting in increased transport of radioactive gases from the main steam supply into the HPCI room.

10CFR55: **41.7**

Technical References: **TS 3.5.1 Bases**
TM-OP-052
M-156 Sht 1

Learning Objectives: **11255 e**

Question Source: **Bank** **ILO LXR TMOP052/2037/007**

Previous NRC Exam: **No**

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Exam	RO	Tier	2	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A		262002 K6.02 Uninterruptable Power Supply (A.C./D.C.)				Importance		2.8	
Statement		Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : D.C. electrical power							

QUESTION 11

Which one of the following identifies the effect on Vital UPS 1D666(2D666) of a loss of the Division 2 250 VDC bus 1D662(2D662)?

	Unit 1 – 1D666	Unit 2 – 2D666
A.	Transfers to ALTERNATE	Transfers to ALTERNATE
B.	Transfers to ALTERNATE	Remains on PREFERRED
C.	Remains on PREFERRED	Transfers to ALTERNATE
D.	Remains on PREFERRED	Remains on PREFERRED

Proposed Answer B

Applicant References None

Explanation The Vital UPS inverter 1D666 is supplied from Class 1E 250V DC bus 1D662. The Unit 2 Vital UPS inverter, 2D666, is supplied from a separate non-Class 1E 250V DC battery, 2D142.

A Incorrect. Unit 2 Vital UPS is powered from 2D142.

B Correct. The 1D666 static switch will automatically transfer to the ALTERNATE supply on undervoltage. 2D666 remains on the PREFERRED source as its supply is unaffected by 2D662.

C Incorrect. This choice represents misapplication of the unit difference to Unit 2.

D Incorrect. While 2D666 remains on the preferred source, 1D2666 does not. This is a plausible distractor as the Computer UPS 1D656 is supplied from the Division 1 250 VDC bus.

10CFR55 41.4

Technical References ON-1(2)88-001
TM-OP-017

Learning Objectives 10174

Question Source New

Previous NRC Exam No

Comments

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Facility Representative ____/______
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Exam	RO	Tier	2	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A	217000 K6.04 Reactor Core Isolation Cooling System					Importance	3.5		
Statement	Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Condensate storage and transfer system								

QUESTION 12

Unit 1 is operating at rated power.

An unisolable leak develops on the RCIC suction line from the Condensate Storage Tank.

Annunciator RCIC CONDENSATE STORAGE LOW LEVEL (AR-108-E01) is in alarm.

Which of the following actions will occur?

A. No actions will occur until the CST level lowers to 36 inches

B. RCIC pump suction from the Suppression Pool, HV-149-F031 OPENS
AND simultaneously
RCIC pump suction from the CST, HV-149-F010 CLOSES

C. RCIC pump suction from the Suppression Pool, HV-149-F031 OPENS
THEN
RCIC pump suction from the CST, HV-149-F010 CLOSES

D. RCIC pump suction from the Suppression Pool, HV-149-F031 OPENS
THEN
RCIC pump suction from the CST, HV-149-F010 CLOSES which will require the operator to manually override and reopen the CST suction valve.

Proposed Answer: C

Applicant References: None

Explanation

A **INCORRECT. The Suppression pool suction valve will automatically begin to stroke open on with 43.5 inches in the CST (alarm setpoint). Additionally the CST suction valve will begin to close automatically when the suppression pool suction valve is full open.**

B **INCORRECT. The suction valve for the suppression pool, HV-149-F031 will open fully. When the valve is full open, a limit switch on the valve will operate a relay contact in the automatic close logic circuit of the CST suction valve, HV-149-F010 to initiate valve closure and to prevent having both the CST and suppression pool suction valves from being open simultaneously.**

C **CORRECT. At 43.5 inches, the RCIC pump suction valve from the suppression pool will begin to open. When the suppression pool suction valve is full open, a limit switch on the valve will operate a relay contact in the automatic close logic circuit of the CST suction valve, HV-149-F010 to initiate valve closure and to prevent having both the CST and suppression pool suction valves from being open simultaneously.**

D INCORRECT. The manual override of the CST suction valve is not required in this condition. This is generally performed during a station blackout (in accordance with EO-100-030) where suppression pool temperatures are elevated and will cause RCIC lube oil to break down

10CFR55	41.7
Technical References	OP-150-001 section 2.2 EO-100-030 Att A
Learning Objectives	11244
Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer mj / 05/16/14
Init / date

Facility Representative ____/______
Init / date

Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	3
K/A	400000 A1.01 Component Cooling Water System					Importance	2.8		
Statement	Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: CCW flow rate								

QUESTION 13

Both units are operating at rated power.

SO-054-A03, Quarterly ESW Flow Verification Loop A, is in progress, with ESW Pump A and C running.

Which one of the following ESW loads, if isolated, would require securing an ESW Pump to avoid pump damage due to potential overheating?

A. Unit 1 OR Unit 2 Reactor Building

B. Any Diesel Generator aligned for standby service

C. 2 or more Diesel Generators aligned for standby service

D. BOTH Control Structure Chillers

Proposed Answer: C

Applicant References: None

Explanation: ESW minimum flow requirements are normally maintained by having the flow paths for all loads valved in. Having both pumps running in a loop requires consideration of pump minimum flow only if more than 1 large load is isolated, per OP-054-001 Step 2.1.2.d. Large loads are defined as either Unit 1 or 2 Reactor Buildings or any Diesel Generator.

A Incorrect. This is only 1 large load.

B Incorrect. This is only 1 large load.

C Correct. ESW isolated to 2 or more DG aligned for standby service constitutes more than 1 large load per Step 2.1.2.d of OP-054-001.

D Incorrect. This is the 2nd largest individual ESW load that can be valved in.

10CFR55: 41.8

Technical References: OP-054-001 Step 2.1.2.d, Att A

Learning Objectives: 10812

Question Source: New

Previous NRC Exam: No

Comments

Operations Reviewer MJ / 6/23/14
Init / date

Facility Representative ____/______
Init / date

Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A	203000 A1.09 RHR/LPCI: Injection Mode					Importance	2.9		
Statement	Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Component cooling water systems								

QUESTION 14

Both units are operating at rated power.

A spurious initiation of Unit 1 RHR Loop B occurs due to a fault in the manual initiation pushbutton.

Which one of the following identifies the RHR Pumps running on Unit 1, and which pump motor oil coolers have cooling water from ESW?

	RHR Pumps running	RHR Pumps with ESW cooling
A.	All	All
B.	All	RHR Pumps 1B, 1C, 1D
C.	RHR Pumps 1B, 1D	RHR Pumps 1B, 1C, 1D
D.	RHR Pumps 1B, 1D	None

Proposed Answer: **A**

Applicant References: **None**

Explanation: **A LPCI initiation signal has been received on Unit 1 Division 2 RHR. Due to the cross-divisional initiation logic, this is equivalent to a full initiation signal to both divisions of LPCI. All 4 RHR pumps receive a start signal and started after their respective time delays. Diesel Generators C and D receive start signals from the divisional LPCI logic. The start of DG C and D will result in starts of the associated C and D ESW Pumps.**

A Correct. All 4 RHR pumps are running on Unit 1 as a result of the Div 2 LPCI initiation. With ESW C and D running ESW is being supplied to all 4 RHR Pump oil coolers.

B Incorrect. This represents an assumption that the DG start signal comes from the respective divisional LPCI logic, and reflects that the RHR Pump 1C oil cooler is cooled from both ESW loops, so that RHR Pump 1C oil cooler receives cooling from ESW B.

C Incorrect. All 4 RHR pumps will be running due to the cross-divisional initiation logic. This choice does reflect that the RHR Pump 1C oil cooler is cooled from both ESW loops.

D Incorrect. All 4 RHR pumps will be running due to the cross-divisional initiation logic, and both loops of ESW will have at least 1 pump running to supply cooling.

10CFR55: **41.7**

Technical References: **M-111 Sht 2, 3**
M1-E11-66 Sht 4
M1-E21-20 Sht 3
TM-OP-054

Learning Objectives: **10805 h**

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Question Source	**Bank**	**ILO LXR TMOP049/181/22**
Previous NRC Exam	**No**	
Comments		

Operations Reviewer MJ / 03JUN14
Init / date

Facility Representative ____/______
Init / date

Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	4
K/A	223002 A2.05 Primary Containment Isolation System/Nuclear Steam SupplyShut-Off					Importance		3.3	
Statement	Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Nuclear boiler instrumentation failures								

QUESTION 15

Unit 1 is operating at rated power.

While I&C is restoring from a channel calibration, ALL Wide Range level indications on the 1C004 panel momentarily lower, to approximately -50", then return to normal.

Which one of the following identifies one effect of the transient, and the operator action required in response?

A. Both Reactor Recirculation Pumps trip
Immediately place the Mode switch to SHUTDOWN

B. RBCW is isolated to the Recirc Pump Motor Coolers
Reset the NSSSS and RBCW isolation logics and reopen the RBCW supply valves

C. RBCCW is isolated to the Recirc Pump Motor Coolers
Reset the NSSSS and RBCW isolation logics and reopen the RBCCW supply valves

D. RBCW is isolated to the Drywell Coolers
Fully open the RBCCW TCV to maximize Drywell cooling

Proposed Answer: **B**

Applicant References: **None**

Explanation: **ON-145-004 Table 2 shows the Wide Range level indications located on the 1C004 panel. A momentary spike to -50" will result in a Level 2 trip at -38".**

A **Incorrect. There are 2 possible methods of tripping the recirc pumps on the -38" signal. ATWS-RPT trips the recirc pumps at -38", but the logic is A+C or B+D to trip the respective trip systems. The A and B channels of the N025 level instruments are affected. The 2nd possible method is due to loss of cooling, but manual action is required there are no automatic trips of the recirc pumps on high pump motor temperature. The M-G set motors do have a direct high motor temperature trip.**

B **Correct. The trip of the A and B channels of the N026 level instruments will result in isolation of RBCW to the Recirc Pump motor coolers via the NSSSS -38" isolation logic. The NSSSS and RBCW isolation logics must be reset and the valves reopened to restore cooling.**

C **Incorrect. RBCCW is supplied to the Recirc Pump bearing and seal coolers, not the motor coolers. RBCCW isolates to the Drywell on a Level 1 isolation signal.**

D **Incorrect. RBCW does isolate to the Drywell coolers, and the specified malfunction would satisfy the logic, but the setpoint is Level 1 (-129").**

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

10CFR55	**41.7**
Technical References	**ON-145-004 Table 2, ON-159-002 Att B** **E-184 Sht 1** **E-216 Sht 11, 29** **M1-B21-131 Sht 7, 10** **TM-OP-059B, TM-OP-080**
Learning Objectives	**11307 h**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer MJ / 03JUN14
Init / date

Facility Representative ____/______
Init / date

Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A		**215005 A2.02 Average Power Range Monitor/Local Power Range Monitor**				Importance		**3.6**	
Statement		**Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale or downscale trips**							

QUESTION 16

Unit 1 is shutting down for a planned outage. Reactor power is 12 percent.

Improved BPWS Control Rod Insertion is being used.

Insertion of a number of high-worth control rods results in a rapid power reduction. Reactor power is 5 percent when control rod insertion is halted.

Which one of the following identifies the <u>next</u> action to be performed, and why?

A. Continue inserting control rods per the shutdown sequence
An unrecognized re-criticality can occur if control rod insertion is stopped

B. Withdraw control rods to raise core power to approximately 10 percent
Reactor power is too low for operation with the Mode switch in RUN

C. Place the Mode switch to SHUTDOWN
Unrecognized re-criticality can occur and continued control rod insertion is blocked

D. Place the Mode switch to STARTUP
Clear the control rod withdrawal block by the APRMs

Proposed Answer **D**

Applicant References **None**

Explanation **With reactor power initially at 12 percent, power is too high to have placed the Mode switch in STARTUP. Per GO-100-004 Step 5.33.9 the Mode switch is not placed to STARTUP until approximately 10 percent power. The next step required by the GO will be to place the Mode switch in STARTUP to clear the APRM downscale control rod withdrawal block at 5 percent power.**

A **Incorrect. Un-recognized criticality does not become a concern until power is less than 3 percent or if subcriticality is confirmed.**

B **Incorrect. Control rod withdrawal is blocked by the APRM downscale at 5 percent.**

C **Incorrect. Un-recognized criticality does not become a concern until power is less than 3 percent. Control rod insertion is not blocked, the APRMs are only generating a withdrawal block, the RWM is bypassed for Improved BPWS.**

D **Correct. The APRMs are generating a control rod withdrawal block that can only be cleared by placing the Mode switch to STARTUP.**

10CFR55 **41.6**

SUSQUEHANNA STEAM ELECTRIC STATION
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REACTOR OPERATOR WRITTEN EXAMINATION

Technical References	**GO-100-004 Step 5.33** **AR-104-H03**
Learning Objectives	**15716**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer MO / 04/23/14
Init / date

Facility Representative ____/______
Init / date

Exam	RO	Tier	2	Group	1	Cognitive Level	Low	Level of Difficulty	3
K/A		264000 A3.04 Emergency Generators (Diesel/Jet)				Importance		3.1	
Statement		Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Operation of the governor control system on frequency and voltage control							

QUESTION 17

Diesel Generator A was being tested at rated load when it tripped due to a spurious high vibration condition.

Diesel Generator trips have been reset per ON-024-001, Diesel Generator Trip.

The Auto Voltage Regulator has NOT been adjusted since the DG tripped.

A test run of the DG is to be performed to demonstrate operability, syncing to ESS Bus 1A.

Which one of the following describes the Control Room indication expected to be observed if the DG is started for the test run without adjusting the Auto Voltage Regulator?

A. Diesel Generator low-priority trouble alarm
DG A volts steady at nominal 4KV

B. Diesel Generator low-priority trouble alarm
DG A volts steady at approximately 4.5KV

C. Diesel Generator high-priority trouble alarm
DG A volts steady at nominal 4KV

D. Diesel Generator high-priority trouble alarm
DG A volts at 0 KV

Proposed Answer: **D**

Applicant References: **None**

Explanation:

ON-024-001 for resetting a DG trip contains a requirement to run the auto voltage regulator setpoint to minimum when resetting a DG trip in preparation for a retest of the engine. This ensure the minimum field current and terminal voltage on the restart. Voltage regulator setup will take place as part of a test run during generator synch and loading/unloading.

Without adjustment of the voltage regulator following a DG trip from full load, an overvoltage trip is expected on subsequent restart of the engine.

A **Incorrect. This describes operation of the DG as for a normal trip reset.**

B **Incorrect. This describes continued operation of the DG with elevated voltage, as expected for a change in generator field.**

C **Incorrect. The overvoltage trip will result in a high-priority DG alarm and trip of the DG.**

D **Correct. An overvoltage trip will generate a high-priority DG alarm and the DG will trip. Voltage indication goes to 0 on a overvoltage trip.**

10CFR55

Technical References	**ON-024-001, Step 3.9, 5.0** **LA-0521-B06, AR-015-B10**
Learning Objectives	**11273 f**
Question Source	**New**
Previous NRC Exam	**No**
Comments	**The K/A was interpreted to include the voltage regulator in addition to the governor due to the failure to reference voltage regulation in the A3 K/A and the importance of the tested concept at SSES.**

Operations Reviewer MD / 03Jun14 **Facility Representative** ____/______
Init / date Init / date

Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A		262001 A3.02 A.C. Electrical Distribution				Importance		3.2	
Statement		Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Automatic bus transfer							

QUESTION 18

Refer to the control panel mimic on the following page when answering this question.

Unit 1 is operating at rated power, Unit 2 is shutting down, in Mode 2.

An electrical transient occurs.

No operator action occurred after the transient.

The final electric plant lineup is shown on the illustration on the following page.

Which one of the following correctly describes the events that led to the electric plant lineup shown?

A. Startup Bus 20 experienced a lockout condition

B. Transformer T-20 experienced a lockout condition

C. Startup Bus 20 breaker to Tie Bus 0A107, 0A104-03, tripped when Tie Breaker 0A105-02 closed

D. Startup Bus 20 feeder breakers tripped on overcurrent when Aux Bus 12B was transferred to Tie Bus 0A107

[Attach sim panel mimic display]

Proposed Answer: **A**

Applicant References: **None**

Explanation: **The electric plant lineup show is that obtained following a Startup Bus 20 lockout, when starting in the normal lineup with the Unit 2 Main Generator offline and the Unit 2 Aux Buses transferred to the Tie Bus.**

A **Correct. On the SUB20 lockout, the feeder breaker from T-20, 0A104-01, and the SUB20 feeder to Tie Bus 0A107, 0A104-03, open. The de-energization of Tie Bus 0A107 initiates a closure signal to the Tie Breaker, 0A015-02 to close. The Tie Breaker permissive to close is met as the Unit 2 Aux Bus 12A and 12B feeder breakers are closed but 0A104-03 is open. The Tie Breaker closes, re-energizing Tie Bus 0A107 and the Unit 2 Aux Buses.**

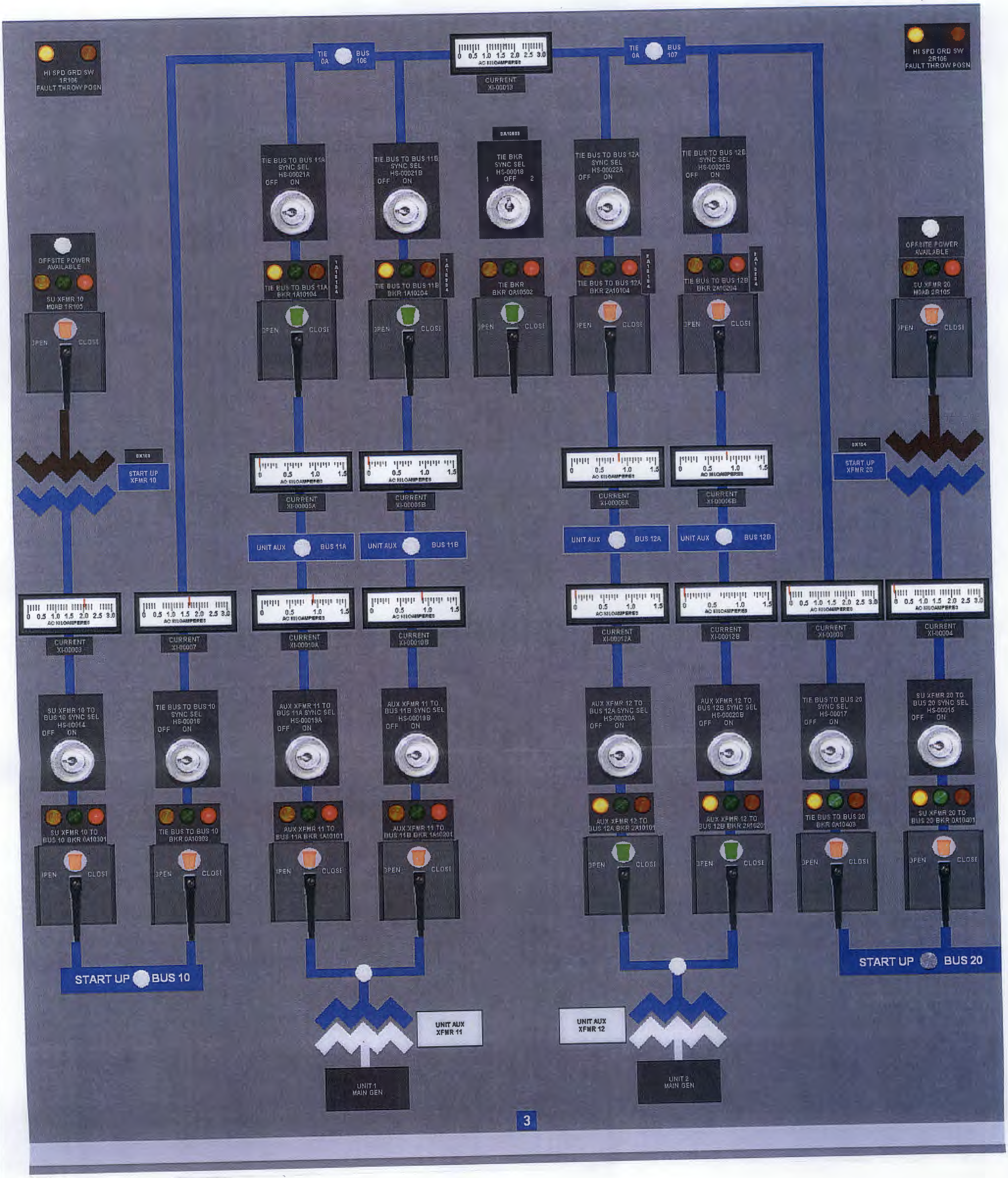
B **Incorrect. On a T-20 lockout the SUB20 breaker to Tie Bus 0A107, 0A104-03, remains closed. 0A104-01 opens, as well as MOAB 2R105. High Speed Ground Switch 2R106 closes.**

C **Incorrect. This distractor is plausible as the Tie Bus auto-closure permissive is that 0A104-03 be open. This distractor represents translation of this starting permissive into an automatic action on an attempted closure of the Tie Breaker. The Tie Breaker would remain open and no automatic closure signal would be generated, with a Unit 2 Aux Bus fed from 0A107 and 0A104-03 closed.**

D **Incorrect. This distractor is plausible for a bus overcurrent condition. However the manual transfer of the Aux Buses was completed successfully, as indicated by the matched semaphores on both Aux Bus feeders from the Unit 2 Main Generator, 2A101-01 and -02.**

10CFR55	**41.7**
Technical References	**ON-003-002 Step 2.10**
Learning Objectives	**11779 I**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer MJ / 032414 **Facility Representative** ____/______
Init / date **Init / date**


HI SPD GRD SW 1R106 FAULT THROW POSN
TIE 0A
BUS 106
CURRENT XI-00013
TIE 0A
BUS 107
HI SPD GRD SW 2R106 FAULT THROW POSN
TIE BUS TO BUS 11A SYNC SEL HS-00021A
TIE BUS TO BUS 11B SYNC SEL HS-00021B
TIE BKR SYNC SEL HS-00018
TIE BUS TO BUS 12A SYNC SEL HS-00022A
TIE BUS TO BUS 12B SYNC SEL HS-00022B
OFFSITE POWER AVAILABLE
SU XFMR 10
TIE BUS TO BUS 11A BKR 1A10104
TIE BUS TO BUS 11B BKR 1A10204
TIE BKR BKR 0A10502
TIE BUS TO BUS 12A BKR 2A10104
TIE BUS TO BUS 12B BKR 2A10204
SU XFMR 20
OPEN
CLOSE
START UP XFMR 10
START UP XFMR 20
CURRENT XI-00005A
CURRENT XI-00005B
CURRENT XI-00006A
CURRENT XI-00006B
UNIT AUX
BUS 11A
BUS 11B
BUS 12A
BUS 12B
AC KILOAMPERES
CURRENT XI-00003
CURRENT XI-00007
CURRENT XI-00010A
CURRENT XI-00010B
CURRENT XI-00012A
CURRENT XI-00012B
CURRENT XI-00004
SU XFMR 10 TO BUS 10 SYNC SEL HS-00014
TIE BUS TO BUS 10 SYNC SEL HS-00016
AUX XFMR 11 TO BUS 11A SYNC SEL HS-00019A
AUX XFMR 11 TO BUS 11B SYNC SEL HS-00019B
AUX XFMR 12 TO BUS 12A SYNC SEL HS-00020A
AUX XFMR 12 TO BUS 12B SYNC SEL HS-00020B
TIE BUS TO BUS 20 SYNC SEL HS-00017
SU XFMR 20 TO BUS 20 SYNC SEL HS-00015
OFF
ON
SU XFMR 10 TO BUS 10 BKR 0A10301
TIE BUS TO BUS 10 BKR 0A10303
AUX XFMR 11 TO BUS 11A BKR 1A10101
AUX XFMR 11 TO BUS 11B BKR 1A10201
AUX XFMR 12 TO BUS 12A BKR 2A10101
AUX XFMR 12 TO BUS 12B BKR 2A10201
TIE BUS TO BUS 20 BKR 0A10403
SU XFMR 20 TO BUS 20 BKR 0A10401
START UP
BUS 10
BUS 20
UNIT AUX XFMR 11
UNIT AUX XFMR 12
UNIT 1 MAIN GEN
UNIT 2 MAIN GEN
3

Exam	RO	Tier	2	Group	1	Cognitive Level	Low	Level of Difficulty	4
K/A	**300000 A4.01 Instrument Air System (IAS)**					Importance		**2.6**	
Statement	**Ability to manually operate and / or monitor in the control room: Pressure gauges**								

QUESTION 19

Two indications of Instrument Air pressure are provided on 1C668 in the Control Room:

PI-12511A, INSTR AIR PRESS
PI-12564, INSTR AIR HDR PRESS

Which one of the following identifies the indications that most closely correspond to

(1) the pressure at which Instrument Air compressor loading is controlled?

(2) the pressure at which the Service Air cross-tie will open?

	Compressor Loading	Service Air Cross-Tie
A.	INSTR AIR HDR PRESS	INSTR AIR HDR PRESS
B.	INSTR AIR HDR PRESS	INSTR AIR PRESS
C.	INSTR AIR PRESS	INSTR AIR PRESS
D.	INSTR AIR PRESS	INSTR AIR HDR PRESS

Proposed Answer **B**

Applicant References **None**

Explanation **The Control Room is provided with 2 indications of Instrument Air pressure for each unit.**

Compressor loading is controlled by the PSL-12508x series of pressure switches. These sense Instrument Air pressure downstream of the Instrument Air Dryers. The INSTR AIR HDR PRESS from PI-1(2)2564 is sensed in the Turbine Building instrument air header.

Service Air cross-tie from PCV-12560 connects to Instrument Air immediately downstream of the Instrument Air receivers. This is where the INSTR AIR PRESS from PI-1(2)2511A is sensed.

A **Incorrect. The pressure at which the S/A cross-tie will open most closely corresponds to INSTR AIR PRESS.**

B **Correct. The I/A compressors are controlled by I/A pressure downstream of the I/A Dryers. This most closely corresponds to INSTR AIR HDR PRESS. The pressure at which the S/A cross-tie will open most closely corresponds to INSTR AIR PRESS.**

C **Incorrect. The I/A compressors operating pressure most closely corresponds to INSTR AIR HDR PRESS.**

D **Incorrect. The I/A compressors operating pressure most closely corresponds to INSTR AIR HDR PRESS. The pressure at which the S/A cross-tie will open most closely corresponds to INSTR AIR PRESS.**

10CFR55 **41.4**

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical References	**M-125 Sht 1,2,3,20**
Learning Objectives	**10588 b**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer MJ / 06/23/14
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Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		261000 A4.04 Standby Gas Treatment System				Importance		3.3	
Statement		Ability to manually operate and/or monitor in the control room: Primary containment pressure							

QUESTION 20

Unit 1 is starting up from a forced outage.

Suppression Chamber inerting is in-progress using Standby Gas Treatment System A.

HD-17508A, DRWL/WETWELL BURP DMP, fails closed.

Which one of the following identifies...

(1) the effect of the damper closure if no operator action is taken?

(2) the appropriate operator action to initiate in response to the failure?

A. Primary containment pressure will rise until the reactor scrams on high Drywell pressure
Terminate the purge by closing HV-15721, CONTN N2 PURGE OB ISO

B. Primary containment pressure will rise until the reactor scrams on high Drywell pressure
Place SGTS B in-service and open HD-17508B, DRWL/WETWELL BURP DMP

C. Primary containment pressure will rise until Drywell pressure reaches 1 psig
Terminate the purge by closing HV-15721, CONTN N2 PURGE OB ISO

D. Primary containment pressure will rise until Suppression Chamber pressure reaches 1 psig
Place SGTS B in-service and open HD-17508B, DRWL/WETWELL BURP DMP

Proposed Answer: C

Applicant References: None

Explanation: A N2 purge of the Suppression Chamber is in progress. The SC is being vented to the common SGTS suction by the HD-17508A and B dampers in series. When the HD-17508A fails closed, venting of the SC via SGTS is no longer possible and SC pressure will begin to rise. When SC pressure is 0.5 psig above Drywell pressure, the DW vacuum reliefs will lift, allowing the SC to vent to the DW and raising DW pressure. SC chamber pressure will continue to rise as long as the N2 supply path is open, so DW pressure will rise, lagging SC pressure by approximately 0.5 psig.

A Incorrect. When Drywell pressure reaches 1 psig the N2 purge supply isolation valve, HV-15721, will automatically close. With the vent path isolated by the HD-17508A failure, SC and DW pressure will remain constant, with the DW at approximately 1 psig, well below the 1.72 psig scram setpoint.

B Incorrect. As noted Drywell pressure will not exceed 1 psig. The HD-17508A is in series with the HD-17508B. The lineup is not 1 valve to each SGTS train.

C Correct. When Drywell pressure reaches 1 psig the N2 purge supply isolation valve, HV-15721, will automatically close. With the vent path isolated by the HD-17508A failure, SC and DW pressure will remain constant, with the DW at approximately 1 psig, well below the 1.72 psig scram setpoint. The containment pressurization transient may be terminated by closing the N2 makeup valve HV-15721 (refer to AR-112-D03).

D Incorrect. The HD-17508A is in series with the HD-17508B. The lineup is not 1 valve to each SGTS train.

10CFR55	41.9
Technical References	OP-173-001 Section 2.1 AR-112-D03 M-157 Sht 1 V-175 Sht 29, E-192 Sht 19 TM-OP-070
Learning Objectives	11181
Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer mw / 03JUN2014
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Facility Representative ____/______
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SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A		**209001 2.4.46 Low Pressure Core Spray System**				Importance		4.2	
Statement		**Emergency Procedures / Plan - Ability to verify that the alarms are consistent with the plant conditions.**							

QUESTION 21

Unit 1 experienced an unisolable steam leak in the RCIC room.

The Unit 1 Aux Buses failed to transfer when the reactor was manually scrammed.

Subsequently, a Rapid Depressurization has been performed due to high temperatures in the HPCI and RCIC rooms.

An operator was directed to perform a component-by-component start of Core Spray Loop A to restore and maintain reactor water level.

Core Spray Pumps 1A and 1C were started.

When the handswitch for HV-152-F005A, CORE SPRAY LOOP A IB INJ SHUTOFF, was placed to OPEN, the valve did not respond.

No other operator action was taken.

The only annunciator associated with Core Spray Loop A in alarm is RHR INJ PERMISSIVE LOOP A RX LO PRESS (AR-109-A05).

Which one of the following describes the preferred method to open HV-152-F005A and inject with Core Spray Loop A under these conditions?

A. Ensure 45 seconds have elapsed since AR-109-A05 went into alarm, THEN open HV-152-F005A using the Control Room handswitch

B. Arm and depress the CORE SPRAY LOOP A MAN INIT pushbutton

C. Place LO RX PRESS PERM on the 1C601 Core Spray Loop A control panel to BYPASS, THEN open HV-152-F005A using the Control Room handswitch

D. Dispatch NPOs to locally open the HV-152-F005A manually

Proposed Answer **C**

Applicant References **None**

Explanation **Following a Rapid Depressurization with a loss of Condensate, reactor level will be low with HPCI and RCIC isolated on low reactor pressure and unavailable to restore reactor level. The conditions presented in the stem stipulate that reactor pressure has fallen below the ECCS low-pressure injection permissive, but reactor level has not lowered to the ECCS automatic initiation setpoint as alarms AR-109-B02 (CS A actuated), -B03 (ECCS hi DW press(and -B04 (ECCS low reactor level) are all clear.**

A Incorrect. There is no time-delay associated with opening low-pressure ECCS injection valves once the low reactor pressure permissive is reached. The 45-sec TD used in this distractor is the TD for manually overriding ECCS injection valves CLOSED once pressure is below the low-pressure permissive and an initiation signal is present.

B Incorrect. Arming and depressing the CS A manual initiation pushbutton is not preferred, as this action will result in a loss of Drywell cooling and subsequent entry into EO-103. OP-AD-004 Att A, V.A.3 directs the operator to take action to initiate ECCS injection prior to the auto-initiation setpoint. Performance of a component-by-component start of CS satisfies this direction.

C Correct. Placing the App R bypass in service bypasses the F005A interlock with the F004A. An ECCS initiation signal is not present to generate an auto-open signal. OP-151-001 Section 2.3.4 provides direction for operation of the App R bypass.

D Incorrect. Operation from the Control Room is possible.

10CFR55	41.7
Technical References	E-155 Sht 12 OP-151-001 Section 2.3.4 OP-AD-004 Att B, Section V
Learning Objectives	10387 c
Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer mJ / 03J4J014
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Exam	RO	Tier	2	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A	217000 2.4.2 Reactor Core Isolation Cooling					Importance	4.5		
Statement	Emergency Procedures / Plan - Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.								

QUESTION 22

Which one of the following sets of alarms represents the <u>mimimum</u> requirement for entry into EO-100-104?

A. RCIC LEAK DETECTION HI TEMP/HI DIFF TEMP (AR-108-E05), ONLY

B. RCIC LEAK DETECTION LOGIC A HI TEMP (AR-108-F04)
<u>OR</u>
RCIC LEAK DETECTION LOGIC B HI TEMP (AR-108-F05)

C. RCIC LEAK DETECTION LOGIC A HI TEMP (AR-108-F04)
<u>AND</u>
RCIC LEAK DETECTION LOGIC B HI TEMP (AR-108-F05)

D. RCIC LEAK DETECTION HI TEMP/HI DIFF TEMP (AR-108-E05)
<u>AND</u>
RCIC LEAK DETECTION LOGIC A HI TEMP (AR-108-F04)
<u>AND</u>
RCIC LEAK DETECTION LOGIC B HI TEMP (AR-108-F05)

Proposed Answer **A**

Applicant References **None**

Explanation **Entry into EO-000-104 is made on area temperatures, radiation levels or room flooding. The alarms listed for consideration all involve EO entry on room temperature. The EO-104 entry conditions (MAX NORMAL temperatures) are set to the setpoint of the first high temperature alarm for area with steam leak detection, such as the RCIC room. The MAX SAFE temperatures for these areas are set to the isolation setpoint. See EO-104 Table**

A **Correct. This is the alarm received for elevated temperatures is the RCIC equipment room at 120 °F room temperature, 45 °F room ΔT.**

B **Incorrect. These alarms are received when the respective channel of the isolation logic trips. These alarms are not required to be received for EO-104 entry as they do not alarm until the isolation setpoint at the MAX SAFE temperature is reached.**

C **Incorrect. These alarms are received when the respective channel of the isolation logic trips. These alarms are not required to be received for EO-104 entry as they do not alarm until the isolation setpoint at the MAX SAFE temperature is reached.**

D **Incorrect. The LOGIC A(B) alarms are received when the respective channel of the isolation logic trips. These alarms are not required to be received for EO-104 entry as they do not alarm until the isolation setpoint at the MAX SAFE temperature is reached.**

10CFR55 **41.7**

Technical References **EO-000-104**
AR-1(2)08-E05

Learning Objectives **14583**

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Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer MJ / 03242014
Init / date

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Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		206000 K2.02 High Pressure Coolant Injection System				Importance		2.8	
Statement		Knowledge of electrical power supplies to the following: System pumps: BWR-2,3,4							

QUESTION 23

Unit 1 scrammed from rated power due to a loss of Feedwater.

Reactor level is being maintained +20" to +45" using RCIC.

Reactor pressure is being maintained 800-1050 psig with HPCI.

DC panel 1D274 is then de-energized.

Which one of the following describes the effect on HPCI, and any operator action required due to the loss of DC power?

A. HPCI will trip
Maintain reactor pressure using SRVs

B. HPCI will receive an isolation signal and trip, but fail to isolate
Close HV-155-F002, STM SUPPLY IB ISO HV

C. HPCI will remain in pressure control
If HPCI trips on high reactor level, maintain reactor pressure using SRVs

D. HPCI trip logic is defeated
Isolate the HPCI steam supply on any HPCI trip signal

Proposed Answer: **C**

Applicant References: **None**

Explanation: **1D274 is the 250V DC power supply to a number of components, including the HPCI Aux Oil Pump and various system valves.**

A **Incorrect. HPCI trip and control logic is power by 125V DC. None of the components affected by the loss of 250V DC power will result in a HPCI trip.**

B **Incorrect. The HPCI isolation logic is powered by 125V DC. None of the HPCI steam supply isolation valves are powered from 1D274.**

C **Correct. On a Level 8 signal the 125 VDC-powered HPCI trip logic will close the Turbine Steam Supply Valve F001, powered from 1D264. As the HPCI turbine coasts down the loss of oil pressure from the shaft-driven main oil pump, with the AOP unavailable, will prevent re-opening the HPCI turbine stop valve if the trip condition clears.**

D **Incorrect. The HPCI trip logic is powered by 125V DC.**

10CFR55: **41.7**

Technical References: **ON-188-001**
TM-OP-052

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Learning Objectives	**11257 b**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer M / 6/24/14
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Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	4
K/A		262001 K5.02 A.C. Electrical Distribution				Importance		2.6	
Statement		Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: Breaker control							

QUESTION 24

The plant experienced a loss of offsite power.

Diesel Generator A started 1 minute ago, but did NOT load onto either ESS Bus 1A or 2A.

Conditions have deteriorated, such that the plant is now in a Station Blackout.

Which one of the following identifies the operation implications of immediately re-energizing ESS Buses 1A and 2A from the Control Room?

A. Entry into EO-100(200)-030 will NOT be required

B. Diesel Generator A will trip due to loss of cooling after a few minutes

C. Installation of Blue Max to 1D613 and 2D613 is no longer required

D. Diesel Generator A will trip due to an overload condition because pump auto-start timers have timed out

Proposed Answer: **B**

Applicant References: **None**

Explanation: **No ESW pumps are in service to provide cooling to Diesel Generator A. ESW Pump A has a pump start signal present. Due to the breaker configuration, the ESW Pump attempts to start onto a de-energize bus and trips with the start signal present. This actuates the anti-pump logic of the ESW Pump breaker. ESW Pump A will not automatically start when ESS Bus 1A is re-energized and cannot be started manually due to the anti-pump feature. Before the ESS Buses can be re-energized DG A must be shutdown locally, then breaker control power to ESW Pump A must be de-energized, then restored, to reset the anti-pump logic. When the Diesel Generator is restarted locally, the associated ESW pump will auto-start.**

A Incorrect. Immediate entry into EO-100(200)-030 will not be required, but after DG A trips in 8 minutes due to a loss of cooling entry will be required.

B Correct. Runtime of a Diesel Generator loaded without cooling is approximately 4.5 minutes, unloaded 8 minutes.

C Incorrect. While the battery chargers will be momentarily restored when ESS Buses 1A and 2A are re-energized, the chargers will be lost once D G A trips due to loss of cooling.

D Incorrect. Pump autostart timers for other pumps will still function to prevent an overload trip of the Diesel Generator.

10CFR55: **41.7**

Technical References: **EO-100-030 Step 2.1**

Learning Objectives: **14625**

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Question Source **New**

Previous NRC Exam **No**

Comments

Operations Reviewer mo / 06/23/14
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Exam	RO	Tier	2	Group	1	Cognitive Level	Low	Level of Difficulty	4
K/A	261000 K1.01 Standby Gas Treatment System					Importance	3.4		
Statement	Knowledge of the physical connections and/or cause-effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: Reactor building ventilation system								

QUESTION 25

Unit 1 is operating at rated power when a small LOCA occurs.

Zone 1 and Zone 3 ventilation isolates.

RB RECIRC SYS TO SGTS DMP, HD-07543A, fails to automatically respond on the Zone 3 isolation signal.

Which one of the following specifies the <u>approximate</u> pressure the Standby Gas Treatment system will be capable of establishing in Zones 1 and 3?

A. more positive than 0" wc

B. 0" wc

C. -0.25" wc

D. more negative than -0.40" wc

Proposed Answer: **C**

Applicant References: **None**

Explanation: **The HD=07543A is 1 of 2 parallel dampers that provide a flowpath from the Reactor Building ventilation Recirc system to SGTS. Failure of just 1 damper still provides a suction source for SGTS to be able to drawdown Zones 1 and 3 to the design negative pressure of -0.25" wc.**

A Incorrect. This choice is consistent with the supply to SGTS isolated in conjunction with normal Zone 1 and 3 ventilation isolated, and the secondary containment slowly pressurizing.

B Incorrect. This choice is consistent with initial response of Zone 1 and 3 pressure to the supply to SGTS isolated in conjunction with normal Zone 1 and 3 ventilation isolated.

C Correct. The SGTS system will still be able to take a suction on Zones 1 and 3 and drawdown Zones 1 and 3 to the design pressure.

D Incorrect. This choice represents a failure of a SGTS modulating damper PPD-07554A to modulate to allow SGTS to limit drawdown to the design pressure of -0.25" wc.

10CFR55: **41.8**

Technical References: **M-175 Sht 2**
ON-159-002 Att B
TM-OP-070

Learning Objectives: **11228 f**

Question Source: **New**

Previous NRC Exam: **No**

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Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A	400000 K4.01 Component Cooling Water System					Importance	3.4		
Statement	Knowledge of CCWS design feature(s) and or interlocks which provide for the following: Automatic start of standby pump								

QUESTION 26

Unit 1 is operating at rated power.

Annunciator RBCCW HEAD TANK HI-LO LEVEL (AR-123-E06) is received.

The NPO dispatched to the RBCCW head tank reports NO level in the tank sightglass. Makeup to the head tank is unsuccessful in recovering level.

The following annunciators are then received:

RBCCW PUMPS DISHARGE HEADER LO PRESS (AR-123-E03)
RBCCW HEAT EXCHANGER HEADER LO PRESS (AR-123-E04)

Operators note PI-11308, RBCCW HX DSH PRESS, is fluctuating widely.

Which one of the following identifies the action to be taken in response to this condition?

A. Depress and release the STOP pushbutton for each RBCCW Pump

B. Depress the STOP pushbutton for the STANDY RBCCW Pump
THEN
Depress the STOP pushbutton for the running RBCCW Pump

C. Depress AND hold the STOP pushbutton for both RBCCW Pumps
THEN
Release the STOP pushbuttons

D. Depress AND hold the STOP pushbutton for both RBCCW Pumps
Open the breakers for both RBCCW pumps
Release the STOP pushbuttons

Proposed Answer: **D**

Applicant References: **None**

Explanation: **RBCCW pumps automatically start on a low pump discharge pressure of 61 psig, as indicated by alarm AR-123-E03, regardless of pump status. In this question a leak has occurred somewhere in the RBCCW system as evidenced by the loss of level in the RBCCW head tank, with makeup to the head tank unable to restore level. The RBCCW pumps are cavitating due to the loss of system inventory as evidenced by the low-pressure alarms and the wide fluctuation in system pressure indicated. The action in response to pump cavitation per ON-114-001 Step 3.8.11 is to stop both RBCCW pumps.**

A Incorrect. The pump auto-start logic will restart each pump as soon as the STOP PB is released.

B Incorrect. The pump auto-start logic does not differentiate between running and standby RBCCW pump. The pump auto-start logic will restart each pump as soon as the STOP PB is released.

C Incorrect. There is no interlock in the auto-start logic that looks at the status of both RBCCW pumps to bypass the auto-start on low system pressure.

D Correct. Both RBCCW pumps receive a start signal on low system pressure that is only bypassed by depressing the pump STOP PB. This is the means to shutdown the RBCCW system per

10CFR55	41.4	
Technical References	E-147 Sht 2 ON-114-001 OP-114-001 AR-123-E03	
Learning Objectives	11086 a	
Question Source	Bank	ILO LXR TMOP014/1694/001
Previous NRC Exam	No	
Comments		

Operations Reviewer MJ / 03/4/2014
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Exam	RO	Tier	2	Group	2	Cognitive Level	Low	Level of Difficulty	3
K/A	215002 K1.05 Rod Block Monitor System					Importance	3.0		
Statement	Knowledge of the physical connections and/or cause/effect relationships between ROD BLOCK MONITOR SYSTEM and the following: Four rod display: BWR-3,4,5								

QUESTION 27

The Rod Block Monitor Operator Display Assemblies located above the 4-Rod Display, on the Standby Information Panel, experience a loss of power.

Which one of the following identifies the effect of the loss of the ODAs on the RBM and the APRMs?

A. No control rod withdrawal blocks
No RPS actuation

B. Control rod withdrawal block due to RDCS inoperable
No RPS actuation

C. Control rod withdrawal block due to RBM inoperable
No RPS actuation

D. Control rod withdrawal block due to APRM inoperable
Full RPS actuation

Proposed Answer: A

Applicant References: None

Explanation: The RBM ODAs comprise part of the OEM 4-rod display. The RBM ODAs provide LPRM indication for the 4 LPRM strings surrounding the selected control rod.

The ODAs are not required for RBM or APRM operability. The ODAs are powered from non-Class 1E 120 V Instrument AC 1Y218-014.

A Correct. Loss of the ODAs has no effect on RBM or APRM operability. No control rod block or scram signals are generated.

B Incorrect. The components powered by 1Y218 on the SIP are not required for RDCS operability. The 4-rod display is powered from 1Y219.

C Incorrect. The operability of the RBM is unaffected by the loss of the ODA.

D Incorrect. The operability of both the RBM and APRMs are unaffected by the loss of their ODAs. Any control rod block will not be due to APRM inoperable.

10CFR55: 41.6

Technical References: ON-117-001 Att A
TM-OP-078K

Learning Objectives: 15804

Question Source: New

Previous NRC Exam: No

Comments

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Exam	RO	Tier	2	Group	2	Cognitive Level	High	Level of Difficulty	2
K/A		201001 K2.02 Control Rod Drive Hydraulic System				Importance		3.6	
Statement		Knowledge of electrical power supplies to the following: Scram valve solenoids							

QUESTION 28

Unit 1 is operating at rated power.

Annunciator BACKUP/GROUP PILOT SCRAM SYSTEM A POWER FAILURE (AR-103-C02) is in alarm.

Which one of the following identifies the initial response to a trip of RPS B if the power loss indicated by the alarm affects the

(1) Backup Scram Valves?

(2) Pilot Scram Valves?

	Backup Scram Valves	Pilot Scram Valves
A.	Both Backup Scram Valves remain closed	1 control rod scrams in
B.	Both Backup Scram Valves remain closed	25% of the control rods scram in
C.	Backup Scram Valve B opens to cause a full reactor scram	1 control rod scrams in
D.	Backup Scram Valve B opens to cause a full reactor scram	25% of the control rods scram in

Proposed Answer B

Applicant References None

Explanation The referenced annunciator is generated from a loss of power to either the A backup scram valve or 1 group of RPS A pilot scram valves. Loss of power to a DC-powered backup scram valve results in the valve failing closed. The loss of power to 1 group of RPS A pilot scram valves will result in approximately 25% of the control rods inserting on a trip of RPS B.

A Incorrect. A trip of RPS B alone is insufficient to generate an open signal to the B Backup Scram Valve. A trip of RPS A also is required, but not indicated. More than 1 pilot scram valve is affected in RPS A; this distractor represent a mis-read of the associated electrical schematic as indicating that the power monitoring relay is only monitoring 1 HCU in the group, not all.

B Correct. A trip of RPS B alone is insufficient to generate an open signal to the B Backup Scram Valve. Approximately 25% of the control rods scram in on the trip of RPS B, as 1 group of RPS A pilot scram valve solenoids were already de-energized as indicated by the initial alarm condition.

C Incorrect. A trip of RPS A in addition to RPS B would be required to energize and open the B Backup scram valve. More than 1 pilot scram valve is affected in RPS A.

D Incorrect. A trip of RPS A in addition to RPS B would be required to energize and open the B Backup scram valve. Approximately 25% of the control rods scram in on the trip of RPS B, as 1 group of RPS A pilot scram valve solenoids were already de-energized as indicated by the initial alarm condition.

10CFR55	41.6
Technical References	AR-103-C02 M1-C72-22 Sht 1, 12, 13
Learning Objectives	10071 d,e
Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer mv / 6/23/14
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Exam	RO	Tier	2	Group	2	Cognitive Level	High	Level of Difficulty	4
K/A	**239001 K3.15 Main and Reheat Steam System**					Importance		3.5	
Statement	**Knowledge of the effect that a loss or malfunction of the MAIN AND REHEAT STEAM SYSTEM will have on following: Reactor water level control**								

QUESTION 29

Unit 1 is operating a 65 percent power.

Inboard MSIV HV-141-F022A fails closed.

The unit remains on-line.

Which one of the following describes how Feedwater level control responds to the MSIV closure?

A. Feedwater level control transfers to 1E-CONTROL
Reactor level lowers slightly due to the MSIV closure, then stabilizes at +35"

B. MSL A flow is substituted as approximately 3.5 Mlbm/hr
Total Steam Flow remains selected for input to 3E-CONTROL
Reactor level rises due to the rise in Total Steam Flow, then then stabilizes at +35"

C. MSL A flow is substituted as 0 Mlbm/hr
Total Steam Flow remains selected for input to 3E-CONTROL
Reactor level drops due to the drop in Total Steam Flow, then then stabilizes at +35"

D. MSL A flow is substituted as approximately 0 Mlbm/hr
Turbine 1st Stage Pressure/Flow selected for input to 3E-CONTROL
Reactor level lowers slightly due to the MSIV closure, then stabilizes at +35"

Proposed Answer: **D**

Applicant References: **None**

Explanation: **The plant will remain on-line for a single MSL isolated at reduced power. Steam flow in the isolated line falls to 0 Mlbm/hr. ICS compares each steam line flow to the high median steam flow, in this case the middle value of the 3 steam line flow for the unisolated lines or approximately 3.5 Mlbm/hr. MSL A flow will be substituted, as it exceeds the ±0.75 Mlbm/hr deviation criteria. The average of the remaining 3 MSL flows is used as the substitute value.**

The total steam flow is then recalculated with the substitute value for MSL A and compared to Turbine 1st stage pressure. Use of the average value through 3 MSL flows will result in a total MSL flow well above the actual MSL flow, as total MSL flow is high by 1/3 due to using a substitute value for MSL A instead of the actual value of 0. Total steam flow will fail the validation test of ±2.1 Mlbm/hr difference when compared to Turbine 1st stage pressure/flow. Turbine 1st stage pressure/flow is then used as the input to 3E-CONTROL, and the substitute value for MSL A flow is set to 0 Mlbm/hr.

Reactor level drops slightly on the MSIV closure due to the momentary pressure spike and void collapse. FWLC in 3-E will quickly stabilize level at the setpoint.

A Incorrect. FWLC doesn't swap to 1E control until both the total MSL flow (due to 2 MSL flow inputs bad or unusable) and turbine 1st stage pressure/flow inputs are both unusable. The response of reactor level is what is expected for a transfer to 1E control simultaneous with a MSL isolation.

B Incorrect. This represents a failure to recognize the validation of the total MSL flow will fail due to being one-third higher than actual MSL flow due to the substitution effect. ICS FWLC is steam-flow dominant, so a sudden rise in steam flow will result in a corresponding rise in FW flow. Level will rise by a few inches, then return to the setpoint as the level deviation integrates in the Master Level Controller.

C Incorrect. The substitute value for MSL A flow is approximately 3 Mlbm/hr. Total steam flow would not drop and induce a level transient due to 3E control action.

D Correct. MSL A flow is substituted as described, Turbine 1st stage pressure is selected due to the Total Steam Flow value being approximately 1/3 higher than actual steam flow, and the only level transient is due to the MSIV closure.

10CFR55	41.5	
Technical References	ON-145-001 Section 2.2, Att B	
Learning Objectives	16087	
Question Source	Bank	LXR LOR TMOP045I/16001/002
Previous NRC Exam	No	
Comments		

Operations Reviewer mJ / 05JUN14
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Exam	RO	Tier	2	Group	2	Cognitive Level	Low	Level of Difficulty	2
K/A	286000 K4.07 Fire Protection System					Importance	3.3		
Statement	Knowledge of FIRE PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Maintenance of fire header pressure								

QUESTION 30

An electrical fire causes the loss of the Motor-Driven Fire Pump.

Which one of the following describes the response of the Fire Protection System to maintain fire header pressure?

A. Backup Motor-Driven Fire Pump starts at 95 psig

B. Diesel Engine-Driven Fire Pump starts at 95 psig

C. Diesel Engine-Driven Fire Pump starts at 85 psig

D. Diesel Engine-Driven Fire Pump starts at 85 psig
Backup Diesel Engine-Driven Fire Pump starts at 85 psig

Proposed Answer: **C**

Applicant References: **None**

Explanation: **A fire has occurred and the Motor-Driven Fire Pump has failed. Only the Diesel Engine Driven Fire Pump is available to maintain fire header pressure. Backup Fire Protection is normally isolated from the main Fire Protection header and is unavailable.**

A **Incorrect. This is the starting setpoint of the Motor-Driven Fire Pump, but while the Backup Fire Protection system contains exact duplicates of the Jockey Fire Pump and the Diesel Engine-Driven Fire Pump, there is no Backup Motor-Driven Fire Pump.**

B **Incorrect. This is the starting setpoint of the MDFP, not the DDFP.**

C **Correct. This is the only standby fire pump aligned to the Fire Protection header. The DDFP auto-starts at 85 psig.**

D **Incorrect. Both DDFP (normal and Backup) auto-start at 85 psig, however Backup Fire Protection is not normally aligned for service.**

10CFR55: **41.4**

Technical References: **OP-013-001**

Learning Objectives: **11385 c**

Question Source: **Bank** **ILO LXR TMOP013/2291/003**

Previous NRC Exam: **No**

Comments

Operations Reviewer MJ / 06/24/13
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Exam	RO	Tier	2	Group	2	Cognitive Level	High	Level of Difficulty	4
K/A	**201003 K5.05 Control Rod and Drive Mechanism**					Importance		3.0	
Statement	**Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM : Reverse power effect**								

QUESTION 31

Unit 1 is starting up. Reactor power is approximately 45 percent.

Operators are withdrawing 12 shallow control rods, from position 40 to position 48, per Reactor Engineering direction.

Which one of the following identifies the operational concern associated with these control rod withdrawals?

A. Violation of the MCPR limit due to excessive bottom-peaked power shape

B. Violation of the MCPR limit due to excessive top-peaked power shape

C. Reduction in reactor power due to change in core void distribution

D. Increased RBM rod out blocks due to the effect on A-level LPRMs

Proposed Answer: **C**

Applicant References: **None**

Explanation: **Withdrawal of shallow control rods will result in a change in core void distribution. Insertion of shallow control rods results in reduced void fractions in the 4 bundles in the control cell, resulting in higher bundle power. When the shallow control rods are withdraw void fractions rise in the now-uncontrolled bundles and total core power lowers.**

- A **Incorrect. MCPR limit violations are typically not of concern at low power/low rod-line conditions. MCPR is more limiting for top-peaked power shape.**
- B **Incorrect. While the MCPR limit is more affected by top-peaked power shapes, this control rod pattern adjustment will result in a much more strongly bottom-peaked power shape, not top-peaked.**
- C **Correct. This is an operational concern, anticipating the effect on core power of shallow control rod withdrawal.**
- D **Incorrect. While A-level LPRMs are most strongly affected by the control rod withdrawal, the A-level LPRMs are not used in the RBM.**

10CFR55: **41.5**

Technical References: **SC056A Chapter 5**

Learning Objectives: **SC056A Ch 5 Obj 12**

Question Source: **New**

Previous NRC Exam: **No**

Comments

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Exam	RO	Tier	2	Group	2	Cognitive Level	Low	Level of Difficulty	2
K/A		**204000 K6.05 Reactor Water Cleanup System**				Importance		2.6	
Statement		**Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER CLEANUP SYSTEM : A. C. power**							

QUESTION 32

Unit 2 startup is in progress, in Mode 2 at 50 psig.

CRD Pump 2B is in-service.

Power to Startup Bus 10 is lost.

ESS Bus 2A fails to transfer to its alternate supply, but is re-energized by Diesel Generator A.

Which of the following describes the effect of the power loss on Unit 2 reactor level?

A. Reactor level is <u>rising</u> due to the loss of Main Turbine EHC

B. Reactor level is <u>rising</u> due to the loss of RWCU blowdown

C. Reactor level is <u>falling</u> due to the loss of CRD

D. Reactor level is <u>falling</u> due to the loss of Condensate

Proposed Answer: **B**

Applicant References: **None**

Explanation: **On Unit 2 a loss of SUB10 will result in a momentary loss of ESS Buses 2A and 2C and a loss of RPS 2A. Unit 2 Aux Buses are unaffected as they are supplied from SUB20. The loss of RPS will result in a loss of RWCU due to a partial isolation by the PCIS outboard logic.**

- **A** **Incorrect. Unit 2 Aux Buses are powered from SUB20. This would be the effect on Unit 1 as main turbine shell warming would isolate on the loss of EHC.**
- **B** **Correct. RWCU pumps would trip and RWCU would be isolated from the reactor due to the RPS 2A trip on the ESS Bus 2A transfer to alternate.**
- **C** **Incorrect. CRD Pump 2B is powered from ESS Bus 2D and is unaffected by the transient.**
- **D** **Incorrect. Condensate would be running per GO-200-002, with 1 pump in service, before the loss of SUB 10. The in-service Condensate would continue to operate. CRD is adequate to maintain reactor level at the power level typical for this reactor pressure, so a loss of Condensate will not affect reactor level.**

10CFR55: **41.5**

Technical References: **ON-003-001**
ON-258-001

Learning Objectives: **11085 g**

Question Source: **New**

Previous NRC Exam: **No**

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Comments

Operations Reviewer mj / 06/24/14
Init / date

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Exam	RO	Tier	2	Group	2	Cognitive Level	High	Level of Difficulty	2
K/A	230000 A1.01 RHR/LPCI: Torus/Suppression Pool Spray Mode					Importance	3.8		
Statement	Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE controls including: Suppression chamber pressure								

QUESTION 33

Unit 1 scrammed from rated power on a turbine trip.

After the scram, primary containment pressure begins to rise. Primary containment pressures are as follows:

Drywell pressure	1.9 psig, steady
Suppression Chamber pressure	2.3 psig, up slow

EO-100-103 is entered for high Drywell pressure.

RHR Loop A is placed in Suppression Chamber spray per OP-149-004, RHR Containment Cooling.

HV-151-F027A, SUPP POOL SPRAY CTL, is opened fully when FI-15120A, CONTN SPRAY DIV 1, fails to respond.

FI-E11-1R603A, RHR A/C FLOW, indicates approximately 550 gpm.

Which one of the following describes the expected response of primary containment pressure in these conditions?

A. Drywell pressure remains steady
Suppression Chamber pressure lowers

B. Drywell pressure remains steady
Suppression Chamber continues to rise

C. Drywell pressure begins to lower
Suppression Chamber pressure remains steady

D. Drywell and Suppression Chamber pressure continue to rise

Proposed Answer	A
Applicant References	None
Explanation	The conditions in the stem are consistent with a leaking SRV with a tailpipe rupture in the Suppression Chamber as indicated by Suppression Chamber pressure greater than Drywell Pressure. Drywell pressure is rising intermittently as the DW-SC vacuum breakers cycle at 0.5 psid. RHR Loop A system flow is indicated as 550 gpm, the approximate value for full flow with a full open spray valve. Action to fully open the spray valve on a failed SC spray indicator is from OP-149-004.

A Correct. The SC spray flow will immediately begin to lower SC pressure due to condensation of steam in the SC airspace from the leaking SRV. DW pressure will remain steady due to the loss of DW cooling and SC no longer relieving steam back to the DW through the DW-SC vacuum breakers.

B Incorrect. If SC pressure continues to rise, DW pressure will rise when the differential between the two compartments exceeds 0.5 psid and the vacuum breakers relieve the SC to the DW.

C Incorrect. The RHR system flow indication is indicative of SC spray flow. SC pressure is expected to lower when spraying a SC filled with steam from a leaking SRV tailpipe before DW pressure would lower.

D Incorrect. SC pressure would be expected to fall due to the indication of SC spray flow. DW pressure would not rise any higher once the rise in SC pressure is arrested.

10CFR55	41.5
Technical References	OP-149-004 Section 2.8.2 EO-000-103 Step PC/P-4
Learning Objectives	10771 s
Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer MJ / 03JUN14
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Exam	RO	Tier	2	Group	2	Cognitive Level	High	Level of Difficulty	4
K/A		**226001 A2.06 RHR/LPCI: Containment Spray System Mode**				Importance		**2.8**	
Statement		**Ability to (a) predict the impacts of the following on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: D.C. electrical failures**							

QUESTION 34

Unit 1 is operating at rated power with RHR Loop B out of service for a SOW.

The reactor scrams due to a small LOCA in the Drywell.

EO-100-103 is entered for high Drywell pressure.

RHR Loop A is placed in Suppression Chamber spray per OP-149-004, RHR Containment Cooling.

Before containment pressure reaches the threshold for Drywell spray, annunciator RHR LOOP A OUT OF SERVICE (AR–109–B09) alarms.

The following conditions are observed:

BIS LOOP A RELAY LGC PWR FAILURE (AR-154-A02)	LIT
RHR LOOP A INIT ISO RESET (HS-E11-1S56A)	Extinguished
LOCA ISOLATION MANUAL OVERRIDE (HS-E11-1S17A)	Extinguished

RHR Loop A Drywell spray valves:

DRYWELL SPRAY IB ISO, HV-151-F021A
DRYWELL SPRAY OB ISO, HV-151-F016A

Which one of the following identifies the preferred method to place Drywell spray in service?

A. Open the <u>outboard</u> HV-151-F016A valve from the Control Room
Open the <u>inboard</u> HV-151-F021A valve locally

B. Open the <u>inboard</u> HV-151-F021A valve from the Control Room
Open the <u>outboard</u> HV-151-F016A valve locally

C. Open <u>both</u> RHR Loop A Drywell spray valves locally

D. Open <u>both</u> RHR Loop A Drywell spray valves from the Control Room

Proposed Answer D

Applicant References None

Explanation	A LOCA has occurred. Only RHR Loop A is available. A loss of RHR logic power occurs before DW sprays can be aligned. The loss of logic power results in losing the manual containment cooling override feature in the RHR logic, but it also defeats the automatic close signal to the DW spray valves from the LOCA signal. There is no interlock between the IB and OB DW spray valves, so both valves can be opened from the Control Room.
	A Incorrect. Local valve operations are not required.
	B Incorrect. Local valve operations are not required.
	C Incorrect. Local valve operations are not required.
	D Correct. The valves can be opened from the Control Room.
10CFR55	41.7
Technical References	AR-154-A02 E-153 Sht 95 M1-E11-66 Sht 4, 5
Learning Objectives	10768 b
Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer MJ / 03JUN14
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Exam	RO	Tier	2	Group	2	Cognitive Level	High	Level of Difficulty	3
K/A		271000 A3.01 Offgas System				Importance		3.3	
Statement		Ability to monitor automatic operations of the OFFGAS SYSTEM including: Automatic system isolations							

QUESTION 35

Unit 1 is operating at rated power.

The following alarms are received

UNIT 1 RECOMBINER CCW PUMP DISCHARGE PRESSURE LO (AR-131-A02)
UNIT 1 RECOMBINER CCW PUMP MOTOR TROUBLE (AR-131-A03)

The alarms cannot be cleared.

Which one of the following identifies the effect of the alarms and the action that can be taken in response?

A. Offgas isolation
Swap Unit 1 to the Common Recombiner

B. Offgas isolation
Place the Common GRRCCW Pump in service

C. ARESD Signal (HV-10721, SJAE DSCH ISO closed)
Re-open SJAE suction valves

D. Recombiner shutdown
Reset the Recombiner Shutdown and return the Recombiner to service

Proposed Answer: **A**

Applicant References: **None**

Explanation: **The alarms received will result in an Offgas isolation on low Recombiner condenser cooling water flow due to trip of the Unit 1 GRCCW pump. As the alarms cannot be cleared the Unit 1 Recombiner cannot be returned to service.**

- **A** **Correct. An Offgas isolation will occur on the pump trip, resulting in closure of the SJAE suction valves. The Common Recombiner must be placed in-service to Unit 1 to restore Offgas.**
- **B** **Incorrect. While an Offgas isolation will occur, the Common GRCCW Pump cannot be aligned to the Unit 1 Recombiner.**
- **C** **Incorrect. Closure of the HV-10721 generates an ARESD signal, it does not result from another initiating condition. The SJAE suction valves cannot be reopened until the Common Recombiner is placed in service.**
- **D** **Incorrect. The Recombiner shutdown signal likely would not be received due to the loss of flow in the GRCCW loop. The Uni1 Recombiner shutdown signal willnot reset and stay reset until the Unit 1 GRCCW Pump can be restarted.**

10CFR55: **41.7**

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Technical References	AR-131-A02, A03 ON-143-001	
Learning Objectives	10930 b	
Question Source	Bank	LXR LOR AD045/15304/086
Previous NRC Exam	No	
Comments		

Operations Reviewer mj / 05/14/14
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Exam	RO	Tier	2	Group	2	Cognitive Level	Low	Level of Difficulty	2
K/A		241000 A4.07 Reactor/Turbine Pressure Regulating System				Importance		3.5	
Statement		Ability to manually operate and/or monitor in the control room: Main stop/throttle valves (operation)							

QUESTION 36

When conducting the quarterly surveillance of Turbine Stop Valves (MSV-1,2,3,4) per SO-193-001, Quarterly Turbine Valve Cycling, which one of the following signals will energize the fast-acting solenoid?

A. First 10 percent of valve stroke

B. First 10 seconds of valve stroke

C. Last 10 percent of valve stroke

D. Stop Valve Test Switch opens

Proposed Answer: C

Applicant References: None

Explanation: Per SO-193-001 Step 5.2.5e, the TSV will fast-close once the valve reaches the 90 percent closed position.

A Incorrect. The valve will fast-close over the last 10 percent of valve position

B Incorrect. The fast-close signal is based on valve position, not stroke time.

C Correct. The valve fast-closes for the last 10 percent of valve stroke.

D Incorrect. The valve fast-closes when the Stop Valve Test Switch closes.

10CFR55: 41.7

Technical References: SO-193-001

Learning Objectives: 1658 h

Question Source: Bank ILO LXR TMOP093E/1658/001

Previous NRC Exam: No

Comments

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Exam	RO	Tier	2	Group	2	Cognitive Level	Low	Level of Difficulty	2
K/A		201006 2.2.42 Rod Worth Minimizer System (RWM)				Importance		3.9	
Statement		Ability to recognize system parameters that are entry-level conditions for Technical Specifications.							

QUESTION 37

Which one of the following requires entry into a Technical Specification LCO?

A. 1 channel of EOC-RPT inoperable at 25 percent power, MCPR limits for inoperable EOC-RPT not applied

B. Extraction steam isolated to 1 of the 2 in-service Feedwater heater strings at 20 percent power

C. Rod Block Monitor A bypassed during startup at 15 percent power

D. Rod Worth Minimizer bypassed for plant shutdown at 10 percent power

Proposed Answer **D**

Applicant References **None**

Explanation **The question presents four conditions for evaluation for LCO entry.**

- **A Incorrect. EOC-RPT operability is not required until 26 percent power per TS 3.3.4.1.**
- **B Incorrect. While ON-147-002 requires entry into LCO 3.2.2 with extraction steam isolated to 1 heater string with only 2 heaters in-service, at 20 percent power MCPR limits do not apply per TS 3.2.2.**
- **C Incorrect. RBM operability is not required until 28 percent power per TS 3.3.2.1.**
- **D Correct. The RWM bypassed at 10 percent power does not comply with LCO 3.3.2.1.**

10CFR55 **41.6**

Technical References **TS 3.3.2.1**

Learning Objectives **13426**

Question Source **New**

Previous NRC Exam **No**

Comments

Operations Reviewer NJ / 03JUN14
Init / date

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Exam	RO	Tier	2	Group	2	Cognitive Level	High	Level of Difficulty	3
K/A		290002 2.2.40 Reactor Vessel Internals				Importance		3.4	
Statement		Ability to apply Technical Specifications for a system.							

QUESTION 38

Use your provided references to answer this question.

Unit 1 is preparing to restart Recirc Pump A at power.

Current conditions are as follows:

Reactor power	30 percent
Load-line	58 percent
Steam dome temperature	539 °F
Bottom head drain temperature	509 °F
Recirc Pump A loop temperature	479 °F
Recirc Pump B loop temperature	514 °F
Recirc Pump B loop flow	18,000 gpm

Which one of the following identifies the action required to proceed with the pump start?

A. Raise Recirc Pump B loop flow ≥ 21,320 gpm

B. Insert control rods to lower reactor power ≤ 27 percent

C. Raise Recirc Pump A loop temperature ≥ 489 °F

D. Maintain Recirc Pump A loop temperature ≥ 464 °F

Proposed Answer: D

Applicant References: TS 3.4.10

Explanation: **An application of TS 3.4.10 is required to determine the action required to allow start of an idle Recirc Pump at power. SR 3.4.10.3 and SR 3.4.10.4 specify the limits to apply to satisfy LCO 3.4.10.**

A Incorrect. This represents a mis-application of the note to SR 3.4.10.6 for power increases in SLO. Operation in SLO would be allowed with loops flows > 21,320 gpm. Start of an idle loop is not allowed by OP-164-001 with flows above 19,500 gpm to protect the TRS limit of 50 percent loop flow (21,320 gpm).

B Incorrect. This represents a mis-application of the note to SR 3.4.10.6 for power increases in SLO.

C Incorrect. This is the action required if the idle loop temperature must be within 50 °F of steam dome temperature. The bases for SR 3.4.10.4 allow the use of running loop temperature to be used as the coolant temperature for the SR.

D Correct. The bases for SR 3.4.10.4 allow the use of running loop temperature to be used as the coolant temperature for the SR. This use is reflected in OP-164-002 Step 2.4.27.d(3). This is the lowest temperature allowed in the idle loop to be within 50 °F of the running loop temperature.

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10CFR55	**41.10**
Technical References	**TS 3.4.10** **OP-164-001 Step 2.4.27.d(3)**
Learning Objectives	**13225**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer rv / 03J4014
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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A		295028 EK1.01 High Drywell Temperature				Importance		3.5	
Statement		Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE : Reactor water level measurement							

QUESTION 39

Unit 1 is operating at rated power when a small steam leak occurs in the Drywell.

Operators note the Wide Range level indications on 1C601 recorders UR-14201A(B), RPV PARAMETERS PAM RECORDER.

Which one of the following identifies the operational implications of a steam leak in the vicinity of the condensing chamber for the Wide Range A level indication?

A. Wide Range A will indicate lower than Wide Range B
Use Wide Range A

B. Wide Range A will indicate higher than Wide Range B
Use Wide Range B

C. Wide Range A will fail downscale
Use Wide Range B

D. Wide Range A will fail upscale
Use Wide Range B

Proposed Answer: B

Applicant References: None

Explanation: Elevated temperatures in the area of a level instrument reference leg will result in erroneously high indicated level due to the higher temperature, lower density fluid in the reference leg. With a steam leak in the area of the D004A reference leg, all Wide Range A level indications will indicate higher than Wide Range B.

- A Incorrect. WR A will indicate higher than WR B.
- B Correct. WR A will indicate higher than WR B. WR B should be selected for reactor level control.
- C Incorrect. WR A would not fail downscale, it would indicate higher.
- D Incorrect. WR A would not fail upscale, the instrument is designed to provide accurate reactor level indication during the DBA LOCA.

10CFR55: 41.5

Technical References: ON-145-004
TM-OP-080

Learning Objectives: 1479 i

Question Source: New

Previous NRC Exam: No

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Comments

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Exam	RO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A		700000 AK1.01 Generator Voltage and Electric Grid Disturbances				Importance		3.3	
Statement		Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Definition of terms: volts, watts, amps, VARs, power factor							

QUESTION 40

Both units are operating at rated power with 1 Reactor Recirc Pump in MONITOR mode.

A grid transient occurs.

Margins to the Main Generator Capability Curve are as shown:

Unit 1 -4 MW, down slow
Unit 2 +2 MW, down slow

TCC contacts the Control Room and requests that both units assume a more lagging power factor.

Which one of the following identifies the action to take on both units to satisfy the TCC request while maintaining margin to the Main Generator Capability Curve?

A. Place Reactor Recirc in MANUAL
Lower Auto Voltage Regulator to operate as close as possible to 0 VARs

B. Ensure Reactor Recirc lowers power
Raise Auto Voltage Regulator as allowed by the capability curve

C. Place Reactor Recirc in MANUAL
Raise Auto Voltage Regulator as allowed by the capability curve

D. Ensure Reactor Recirc lowers power
Lower Auto Voltage Regulator to operate as close as possible to 0 VARs

Proposed Answer: B

Applicant References: None

Explanation: Unit 1 is operating above the main generator capability curve, with Unit 2 approaching the curve, due to a grid transient that resulted in both units assuming more reactive load. Action must be taken on Unit 1 to restore operation within the capability curve. TCC has requested both units assume a more lagging power factor. This requires both units to assume more reactive loading and raise vars. The MONITOR mode of recirc will initiate core power reductions to lower MWe loading to restore margin to the capability curve.

A Incorrect. Placing recirc in manual will result in MWe remaining constant. Combined with lowering VARs this will result in operating with a more leading power factor.

B Correct. Lowering MWe will restore margin to the capability curve allowing the units to assume more VAR loading, resulting in a more lagging power factor

C **Incorrect. With MWe constant due to the action to maintain core power constant, the only adjustment possible to the Auto Voltage regulator is to lower VAR loading to restore margin to the capability curve. Lowering VARs results in a more leading power factor.**

D **Incorrect. Lowering VARs will result in a more leading power factor.**

10CFR55	**41.5**
Technical References	**ON-198-001** **TM-OP-098D**
Learning Objectives	**10850 a**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer MJ / 03/21/2014
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Exam	RO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		295021 AK1.04 Loss of Shutdown Cooling				Importance		3.6	
Statement		Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : Natural circulation							

QUESTION 41

Unit 1 is shutting down for a refueling outage in Mode 4, reactor coolant temperature 195 °F.

RHR is operating in Shutdown Cooling.

Recirc Pumps are shutdown.

Reactor head vents to the Drywell sump have <u>NOT</u> been opened.

A leak results in reactor level lowering to +5".

Which one of the following identifies

(1) the correct indication for determining vessel heatup rate?

(2) how entry into Mode 3 would be indicated?

A. RWCU bottom head drain temperature (NLT01 or TR-B21-1R006)
Mode 3 entry is indicated by RWCU drain temperature

B. Reactor vessel skin temperature (TE-B21-1N030E)
Mode 3 entry is indicated by reactor vessel skin temperature

C. Reactor vessel skin temperature (TE-B21-1N030E)
Mode 3 entry must be inferred from steam dome pressure rise

D. No valid coolant temperature indication is available
Mode 3 entry must be inferred from steam dome pressure rise

Proposed Answer: **C**

Applicant References: **None**

Explanation: **With reactor level falling to +5" a RHR SDC isolation occurs. RHR pumps trip on loss of suction path as the RHR F008 and F009 valves close. No recirc pumps are running and level is < 45" so no core coolant circulation is occurring. ON-149-001 specifies the methods for determining vessel heatup. With the reactor head vents not yet aligned to the Drywell sump the reactor will pressurize as coolant temperature rises and reaches saturation in the core.**

A Incorrect. Use of RWCU for coolant temperature is not allowed by ON-149-001 Step 3.4.6.b as no core circulation is occurring. RWCU drain temperature is not indicative of core coolant temperature.

B Incorrect. Although ON-149-001 does allow the use of vessel skin temperature under these conditions, this temperature is not reflective of core coolant temperature because of the lack of circulation in the reactor due to the low reactor level and no recirc pumps running.

C Correct. ON-149-001 does allow the use of vessel skin temperature under these conditions. Entry into Mode 3 will be indicated when reactor pressure begins to rise as core coolant temperature reaches saturation and starts to steam.

D Incorrect. ON-149-001 does allow the use of vessel skin temperature under these conditions.

10CFR55	41.5	
Technical References	ON-149-001 Step 3.5, 5.0	
Learning Objectives	10771 r	
Question Source	Bank	LOR LXR TMOP049/10771/001
Previous NRC Exam	No	
Comments		

Operations Reviewer MW / 06/24/14
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3.4 **Determine** cause of loss of RHR Shutdown Cooling, **AND** **Perform** the following:

☐ 3.4.1 IF conditions permit restoring the previously in-service loop of RHR to Shutdown Cooling, **Perform** Attachment C, Quick Recovery of Previously Inservice Shutdown Cooling Loop.

AND

☐ 3.4.2 IF loss occurred in Mode 3 or Mode 4, **Perform** Section 3.5 of this procedure.

☐ 3.4.3 IF loss occurred in Mode 5 **AND** level < 22 feet above flange, **Perform** Section 3.6 of this procedure.

☐ 3.4.4 IF loss occurred in Mode 5 **AND** level > 22 feet above flange, **Perform** Section 3.7 of this procedure.

3.5 IF RHR Shutdown Cooling lost in Mode 3 or Mode 4:

☐ 3.5.1 IF in Mode 3, **Comply** with TS 3.4.8.

☐ 3.5.2 IF in Mode 4, **Comply** with TS 3.4.9.

☐ 3.5.3 IF in Mode 4, **Review** Attachment G to determine estimated "Time to 200 F."

☐ 3.5.4 IF SDC lost due to Loss of RHRSW, **Restart** RHRSW IAW OP-116/216-001, else N/A.

3.5.5 IF one loop of RHR Shutdown Cooling lost:

a. **Promptly Establish** reactor coolant circulation using **ONE** of the following alternate methods:

☐ (1) **Maintain** water level $\geq$ 45 inches.

☐

NOTE:	Placing Reactor Recirculation System in service will provide accurate indication of Coolant Temperature but will also add heat to the coolant over time.

☐ (2) **Ensure** Reactor Recirculation System in service IAW OP-164-001.

Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A		600000 AK2.01 Plant Fire On Site				Importance		2.6	
Statement		Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors / detectors and valves							

QUESTION 42

The Control Structure HVAC rooms in Area 12/21-783 are protected by a Pre-Action Sprinkler system.

Which one of the following identifies the condition(s) required to be met to discharge fire suppression water into the area in the event of a fire?

A. Simplex Priority 1 alarm must be received for the area

B. Simplex Priority 2 alarm must be received for the area

C. Area temperatures must exceed the melt temperature of the sprinkler head fusible links
AND
Simplex Priority 2 alarm must be received for the area

D. Area temperatures must exceed the melt temperature of the sprinkler head fusible links
AND
OS&Y valve must be opened
AND
Sprinkler system isolation valve must be opened

Proposed Answer: C

Applicant References: None

Explanation: **Pre-action Sprinkler systems require 2 conditions be satisfied to discharge fire suppression water into an area. First the sprinkler piping must be charged by opening the pre-action valve. This occurs when the Simplex Priority 1 alarm is received. Second area temperatures must rise sufficiently to melt the fusible links in the closed sprinkler heads in the area. Only when both conditions are satisfied will fire suppression water be discharged into the area.**

A **Incorrect. This is indicative of opening of the pre-action valve to charge the sprinkler header. No fire suppression water is discharged into the area until the sprinkler fusible heads melt. This distractor is a description of how a pre-action deluge system functions.**

B **Incorrect. This is indicative of fire detection in the area. No action is initiated by the fire suppression system on the detection alarm.**

C **Correct. This is indicative of opening of the pre-action valve to charge the sprinkler header. No fire suppression water is discharged into the area until the sprinkler fusible heads melt.**

D **Incorrect. The local valve operations are required for a manual deluge system.**

10CFR55 **41.4**

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Technical References	**FP-013-186** **OP-013-001 Step 2.5.3.a Note** **AR-SP-002** **AR-SP-001** **TM-OP-013, TM-OP-013Z**
Learning Objectives	**11383 h**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer mo / 03Jun14
Init / date

Facility Representative ____/______
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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A	295005 AK2.02 Main Turbine Generator Trip					Importance	2.9		
Statement	Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Feedwater temperature								

QUESTION 43

Which one of the following identifies why reactor power rises by approximately 4 percent when the Main Turbine is tripped during a normal plant shutdown per GO-1(2)00-004, Plant Shutdown to Minimum Power?

A. Reactor pressure rises to a new steady-state value when pressure control is transferred to the bypass valves

B. Extraction steam is isolated to the #1 and #2 Feedwater heaters, ONLY, when the Main Turbine is tripped

C. Extraction steam is isolated to the #3, #4 and #5 Feedwater heaters, ONLY, when the Main Turbine is tripped

D. Extraction steam is isolated to all Feedwater heaters when the Main Turbine is tripped

Proposed Answer: **D**

Applicant References: **None**

Explanation: **A turbine trip results in isolation of extraction steam to all Feedwater heaters. #5 heater is 9th stage extraction from the HP turbine, all others are extraction steam from the LP turbines. A turbine trip isolates extraction steam to all FW heaters. In GO-1(2)00-004 Step 5.28 Note states reactor power rises approximately 4 percent on the turbine trip.**

- A **Incorrect. Reactor pressure does rise slightly after the trip of the main turbine, approximately 3 psig. This variation in reactor pressure is small and insufficient to cause power to rise by 4 percent.**
- B **Incorrect. Extraction steam is isolated to all FW heaters. This distractor is plausible as the #1 and #2 heaters do not have MOV isolation valves in their extraction steam supply.**
- C **Incorrect. Extraction steam is isolated to all FW heaters. This distractor is plausible as the #3, #4 and #5 heaters all have MOV isolation valves in their extraction steam supply.**
- D **Correct. All extraction steam is isolated to all FW heaters on a turbine trip.**

10CFR55: **41.5**

Technical References: **GO-100-004 Step 5.28 Note**
M-102-1

Learning Objectives: **11172**

Question Source: **New**

Previous NRC Exam: **No**

Comments

Operations Reviewer MW / 03JUN14
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CONFIRM

☐ | NOTE: Reactor power will rise approximately 4% following Turbine trip due to loss of feedwater heating.

5.28 **Perform** following to remove the Turbine/Generator from service (N/A following Rx SCRAM):

5.28.1 Main Turbine Pre-outage Overspeed Test in accordance with OP-193-002, as scheduled. ________

________ RPM
Overspeed Trip

5.28.2 **Shutdown** Main Turbine in accordance with OP-193-001. ________

5.28.3 **Remove** Extraction Steam from all Feedwater Heater Strings in accordance with OP-147-001. ________

5.28.4 **Shutdown** Generator IAW OP-198-001. ________

5.29 **IF** Service Water System heat loads are reduced to where pump dead heading may occur, **Align** Service Water System for a one pump alignment in accordance with the "Shutdown to One Service Water Pump" section of OP-111-001. ________

5.30 **Perform** Attachment 'B', Defeat the Alarm Reflash From Feedwater Heater Panels 1C101, 1C102, and 1C103 To Control Room Panel 1C668. ________

5.31 **Perform** either of the following sections as applicable:

5.31.1 Step 5.32 – **IF** performing a MANUAL SCRAM ________

5.31.2 Step 5.33 – **IF** performing a SOFT SHUTDOWN ________

Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A		295031 EK2.15 Reactor Low Water Level				Importance		3.2	
Statement		Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: A.C. distribution							

QUESTION 44

Unit 1 experiences a hydraulic block ATWS.

Initial ATWS power is 40 percent.

SLC injection is successful.

With reactor power at 10 percent Feedwater is lost.

Reactor level falls to -135".

Which one of the following identifies the loads that will be shed for electrical distribution protection if the Main Turbine trips at this time?

A. TBCCW Pumps

B. Service Water Pumps

C. Instrument Air Compressors

D. Turbine Building Chillers

Proposed Answer	B
Applicant References	None
Explanation	A Main Turbine trip will result in a Main Generator lockout. When the Main Generator lockouts trip with a LOCA initiation signal on low reactor level (-129") sealed-in, the Aux Buses undergo a Plant Aux load shed. Major 13.8 KV loads on the Aux Buses receive a momentary trip signal to ensure the Startup Buses are not overloaded when the Aux Buses fast transfer to the Tie Bus.

A TBCCW pumps are powered from 480V MCCs supplied by the Aux Buses. The power supplies to the TBCCW Pumps are not shed on the Plant Aux Load Shed.

B Correct. Service Water Pumps are shed on the Plant Aux Load Shed.

C Incorrect. Instrument Air Compressors are locked out for 10 minutes on a -129" signal, but only if a Loss of Offsite Power has occurred.

D Incorrect. The TB Chillers are powered from the ESS Buses. This distractor is plausible in that these chillers are shed on the LOCA signal, but are not affected by the status of the main generator.

10CFR55	41.4
Technical References	E-102 Sht 31 E-145 Sht 1
Learning Objectives	11779 h

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LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer MD / 03JUN14
Init / date

Facility Representative ____/____
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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	3
K/A		298016 AK3.03 Control Room Abandonment				Importance		3.5	
Statement		Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT : Disabling control room controls							

QUESTION 45

Which one of the following identifies the reason for disabling the Control Room HMIs during performance of ON-100-009, Control Room Evacuation, for a Control Room fire?

A. Enable manual control of LV-10641, FW STARTUP RX LEVEL CONTROL VLV, at 1C1115

B. Prevent uncontrolled condensate injection by spurious opening of LV-10641, FW STARTUP RX LEVEL CONTROL VLV

C. Prevent uncontrolled condensate injection by spurious re-opening of any HV-10603A(B)(C), RFP DSCH ISO VLV

D. Ensure SETPOINT SETDOWN remains in effect to maintain reactor level as low as possible to avoid RCIC high-level trip

Proposed Answer: C

Applicant References: None

Explanation: In the event of a Control Room fire, ON-100-009 identifies the primary concern with fire-induced misoperation of ICS as re-opening of the HV-10603x RFP discharge valves. Re-opening of these valves during a controlled reactor cooldown would result in uncontrolled condensate injection and vessel flooding. With the 10603 valves remaining closed, as long as the pumps are running Condensate should remain available to inject, and ICS can automatically maintain reactor level via the 10641 startup level control valve.

A Incorrect. The HMI at the 1C1115 panel is provided for observing performance of the LV-10641 valve, not to enable control. The 1C1115 view-only HMI is referenced in the Caution to ON-100-009 Step 4.6.

B Incorrect. The LV-10641 valve is left in AUTO to allow ICS to be able to maintain reactor water level when Condensate Pumps are capable of injection. Spurious re-opening of the 10603x valves is the primary concern of allowing the Control Room HMIs to remain functional during a Control Room fire.

C Correct. Spurious re-opening of the 10603x valves is the primary concern of allowing the Control Room HMIs to remain functional during a Control Room fire.

D Incorrect. Setpoint Setdown is reset as part of the ON-100-009 Immediate Operator Actions to allow ICS to maintain reactor level in the normal band when Condensate Pumps are capable of injection. RCIC high-level trips are defeated when control is transferred to the RSDP.

10CFR55: 41.10

Technical References: ON-100-009

Learning Objectives: 15320

Question Source: New

Previous NRC Exam: No

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Operations Reviewer mj / 05/14/14
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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A	295037 EK3.01 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown				Importance		4.1		
Statement	Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Recirculation pump trip/runback								

QUESTION 46

Which one of the following describes why Recirc Pumps are runback to minimum speed for an ATWS where RPS fails to trip at rated power?

A. Limit development of potentially fuel-damaging power oscillations

B. Anticipation of a Recirc LIM1 runback when reactor level is lowered

C. Reduce dilution of Standby Liquid Control boron by circulation through the recirc piping

D. Prevent containment heatup due to tripping the Main Turbine and exceeding bypass valve capacity

Proposed Answer: D

Applicant References: None

Explanation: In an ATWS at rated power with a failure of RPS to trip, the immediate priority in executing EO-000-113 is to lower reactor power. After SLC is initiated recirc pumps are tripped for a rapid power reduction. Prior to tripping recirc pumps, if any steam-driven turbine is in operation recirc speed is reduced to minimum first. This is to prevent high-level trips of the steam turbine.

A Incorrect. The reduction in recirc flow will actually make the development of large power oscillations more likely. SLC initiation and reactor level reduction are performed in part to compensate for the decrease in stability margin when recirc flow is reduced.

B Incorrect. No concern for anticipating the automatic runback is identified by EO-000-113. The only discussion of a LIM1 runback in Step LQ/Q-7 of EO-000-113 is that action is required to initiate a runback to minimum if a LIM1 runback has not occurred.

C Incorrect. SLC concentration and inventory limits are established full mixing of the boron solution in the recirc loops.

D Correct. Specifically, trip of the main turbine in an ATWS with no rod motion will likely result in power levels exceeding bypass valve capability. As a result, primary containment will be challenged per EO-000-113 Step LQ/Q-7.

10CFR55: 41.5

Technical References: EO-000-113

Learning Objectives: 14613

Question Source: New

Previous NRC Exam: No

Comments

Operations Reviewer MJ / 03JUN14
Init / date

Facility Representative ____/______
Init / date

Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A		**295038 EK3.03 High Off-Site Release Rate**				Importance		**3.7**	
Statement		**Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Control room ventilation isolation**							

QUESTION 47

Which one of the following identifies why the CREOASS system is initiated in the PRESSURIZATION mode in response to a Zone I or II isolation signal?

A. To ensure Control Room equipment OPERABILITY is maintained by providing a controlled environment during accident conditions

B. To ensure Control Room equipment OPERABILITY is maintained by minimizing the intake of radioactive material in the Control Room

C. To minimize dose to Control Room personnel because a LOCA resulting in high on-site and off-site release rates could be in progress

D. To minimize dose to Control Room personnel because a fuel handling accident may have resulted in gross fuel cladding damage

Proposed Answer **C**

Applicant References **None**

Explanation **The applicant is asked to identify the bases for the Control Room isolation function (CREOASS actuation in the Pressurization mode) in response to a Zone 1 or 2 isolation signal. The Zone 1 and 2 isolation signals are reactor level low (-38") and high Drywell pressure (1.72 psig). Initiating CREOASS in the PRESSURIZATION mode isolates the normal Control Room fresh air intake and aligns the intake to the CREOASS filter trains, ensuring the air intake is filtered before release into the Control Room.**

A **Incorrect. This distractor is describing the basis of the Control Room Floor Cooling systems, which maintain a controlled temperature environment in the Control Room in normal and accident conditions.**

B **Incorrect. While CREOASS does limit the intake of radioactive material into the Control Room atmosphere, the purpose of the system as described in TS Bases 3.7.3 is to limit personnel exposure, not ensure equipment remains within the assumed EQ.**

C **Correct. The purpose of the CREOASS initiation is to limit minimize dose to Control Room personnel by aligning the system to provide a source of filtered air if the event in progress degrades into a LOCA.**

D **Incorrect. This distractor is describing the basis of the CREOASS actuation on high secondary containment ventilation radiation levels, which are Zone III isolation signals, not Zone 1 or 2.**

10CFR55 **41.12**

Technical References **TS 3.7.4 Bases**

Learning Objectives **13057**

Question Source **New**

Previous NRC Exam **No**

Comments

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Operations Reviewer mo / 03242014
Init / date

Facility Representative ____/______
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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A		295026 EA1.03 Suppression Pool High Water Temperature				Importance		3.9	
Statement		Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature monitoring							

QUESTION 48

Which one of the following identifies conditions when Suppression Pool temperature cannot be determined?

A. Control has been transferred to the Remote Shutdown Panel

B. Suppression Pool temperature > 230 °F

C. Suppression Pool level < 20.5 ft

D. 1Y216 is de-energized

Proposed Answer: **B**

Applicant References: **None**

Explanation: **Suppression Pool temperature is monitored by a network of 20 RTDs connected to 2 divisionalized SPOTMOS NUMAC panels with additional monitoring capability at the Remote Shutdown Panel. SPOTMOS calculates 3 average temperatures. The SPOTMOS RTDs are located at 20.5' SP level and approximately 3.5' SP level.**

1) **Bulk SP Temp is the preferred indication with SP level > 20.5'. It utilizes both upper and lower RTDs.**
2) **SPOTMOS average temp is available with SP level > 20.5' It utilizes upper RTDs only.**
3) **SPOTMOS bottom-average temp is available for SP level > lower RTD location. It utilizes only lower RTDs.**

A **Incorrect. SP temperature indication is available at the RSDP on TI-15751 once control has been transferred to the RSDP per ON-100-009.**

B **Correct. SP temperature cannot be determined above 230 °F per EO-000-103 Step SP/T-1 bases. This is the upper limit for the RTD indication.**

C **Incorrect. This is the level at which the upper RTDs are located. 2 of the 3 average SP temperature measurements are lost, but the bottom average remains.**

D **Incorrect. All of the lower RTDs are powered from Division 1. This is the alternate power supply for the Division 1 SP temperature RTDs.**

10CFR55: **41.7**

Technical References: **EO-000-103 SP/T-1**
TM-OP-059Z

Learning Objectives: **10507 e**

Question Source: **New**

Previous NRC Exam: **No**

Comments

Operations Reviewer MD / 03JUN14
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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	4
K/A	295004 AA1.01 Partial or Complete Loss of D.C. Power				Importance		3.3		
Statement	Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : D.C. electrical distribution systems								

QUESTION 49

Unit 1 is operating at rated power.

During a routine panel test, the following annunciator panels are found to be non-responsive:

1C651	AR-104	Division 2 RPS
	AR-105	Main Turbine
	AR-106	Main Generator, Electrical
1C668	All annunciators	

Which one of the following identifies the first electrical distribution system to investigate?

A. 1D645, 125V DC

B. 1D662, 250V DC

C. 1D240, Instrument AC UPS

D. 1Y246, Instrument AC

Proposed Answer: A

Applicant References: None

Explanation: The only direct means of monitoring the DC and 120V AC systems in the SSES Control Room is via annunciation on Control Room panels 1C651, on annunciator panel AR-106. Power for AR-106 is from ESS 125 VDC panel 1D645.

A Correct. 1D645 breakers 16 and 19 power the affected annunciator panels.

B Incorrect. This is a plausible choice, in that it is the Division 2 ESS 250 VDC system.

C Incorrect. This is a plausible choice, in that it is the 120V Instrument AC UPS powered from a dedicated battery backup.

D Incorrect. This is a plausible choice, in that it is the Class 1E 120V Instrument AC power for Division 2 from the D load group.

10CFR55: 41.7

Technical References: ON-102-640
AR-106-G17

Learning Objectives: 10983 e

Question Source: New

Previous NRC Exam: No

Comments

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Operations Reviewer MD / 03JUN14
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Exam	RO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		295018 AA1.01 Partial or Complete Loss of Component Cooling Water						Importance	3.3
Statement		Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Backup systems							

QUESTION 50

Which one of the following describes how Drywell cooling is provided following a loss of offsite power?

A. All previously running Drywell cooler fans automatically restart
RBCCW automatically aligns to supply Drywell coolers
ESW must be manually aligned to supply RBCCW heat exchangers

B. All previously running Drywell cooler fans automatically restart
RBCCW must be manually aligned to supply Drywell coolers
ESW automatically aligns to supply RBCCW heat exchangers

C. All Drywell cooler fans automatically restart, except the undervessel and reactor head area coolers
RBCCW automatically aligns to supply Drywell coolers
ESW must be manually aligned to supply RBCCW heat exchangers

D. All Drywell cooler fans automatically restart, except the undervessel and reactor head area coolers
RBCCW must be manually aligned to supply Drywell coolers
ESW automatically aligns to supply RBCCW heat exchangers

Proposed Answer: A

Applicant References: None

Explanation: A loss of offsite power occurred. Drywell coolers all restart when the Diesel Generators re-energize the respective ESS Buses. RBCCW will automatically realign to supply cooling to the Drywell coolers due to a loss of power to both RBCW chilled water pumps. RBCCW heat sink to service water is lost, so RBCCW HX cooling must be realigned from Service Water to ESW.

A Correct. This describes the normal plant response to a loss of offsite power.

B Incorrect. RBCCW automatically aligns to the Drywell coolers, and ESW must be manually aligned to RBCCW.

C Incorrect. The undervessel and reactor head area coolers response differs from the other DW coolers under LOCA, not LOP, conditions.

D Incorrect. The undervessel and reactor head area coolers response differs from the other DW coolers under LOCA, not LOP, conditions. RBCCW automatically aligns to the Drywell coolers, and ESW must be manually aligned to RBCCW.

10CFR55: 41.4

Technical References: ON-104-001
E-224 Sht 1
E-216 Sht 4, 8, 9
TM-OP-073

Learning Objectives: 11191 b

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Question Source	**Bank**	**ILO LXR TMOP073/1882/001**
Previous NRC Exam	**No**	
Comments		

Operations Reviewer MJ / 03JUN14
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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	3
K/A		295030 EA2.03 Low Suppression Pool Water Level				Importance		3.7	
Statement		Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Reactor pressure							

QUESTION 51

Which one of the following identifies when bypassing interlocks and re-opening MSIVs is allowed to lower reactor pressure, in accordance with EO-000-112, Rapid Depressurization?

A. HCTL violated in ATWS, initial ATWS power < 5 percent and SLC not initiated

B. Primary Containment pressure approaching 65 psig

C. Reactor pressure > 95 psig with 4 SRVs open

D. Suppression Pool level < 5 ft

Proposed Answer: D

Applicant References: None

Explanation: The question requires the determination of whether alternate RPV vent paths are required to accomplish Rapid Depressurization per EO-000-112. The question requires the applicant to interpret the effect of low Suppression Pool level when reactor pressure must be reduced via Rapid Depressurization in EOPs to determine the correct answer.

A Incorrect. In an ATWS, bypassing interlocks and re-opening MSIVs is allowed per Step LQ/P-5 of EO-000-113, but only when SLC is required. SLC is not required in this choice. With HCTL violated in a low-power ATWS, Rapid Depressurization is required with SRVs.

B Incorrect. PC pressure approaching 65 psig requires terminating injection into the reactor and PC from external sources via overrides in EO-000-102 and -103. No special direction regarding reactor depressurization is provided.

C Incorrect. Reactor pressure > 95 psig with 4 SRVs open does not allow use of alternate vent paths, per EO-000-112 Step RD-11.

D Correct. Suppression Pool level this low will result in uncovering SRV spargers and direct steam release to the Suppression Chamber airspace. Use of alternate RPV vent paths is required.

10CFR55: 41.10

Technical References: EO-000-112 Step RD-7, 11, 13

Learning Objectives: 14593

Question Source: New

Previous NRC Exam: No

Comments:

Operations Reviewer mj / 5/14/14
Init / date

Facility Representative ____/______
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Exam	RO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	4
K/A		**295001 AA2.05 Partial or Complete Loss of Forced Core Flow Circulation**				Importance		**3.1**	
Statement		**Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Jet pump operability**							

QUESTION 52

Unit 1 experienced a reduction in core power and generator output.

Operators note the following parameter change as shown

Indicated core flow	Higher
Core plate ΔP	Lower
Recirc Pump B flow	Higher
Loop A JP flows	Higher
Loop B JP flows	Lower
Jet pump 9 flow	Lower
Jet pump 10 flow	Higher

Which one of the following identifies

(1) the most likely cause of the observed indications?

(2) whether the jet pumps are still capable of performing their required safety function?

A. Displaced jet pump mixer
Jet pump safety function is NOT maintained

B. Loose jet pump mixer
Jet pump safety function is NOT maintained

C. Loose jet pump mixer
Jet pump 10 is INOPERABLE
Jet pump safety function is maintained for all other jet pumps

D. Plugged jet pump nozzle
Jet pump 9 is INOPERABLE
Jet pump safety function is maintained for all other jet pumps

Proposed Answer A

Applicant References **None**

Explanation	**ON-164-005 provides guidance on diagnosing jet pump failures. The reduction in core power, generator load and core plate ΔP indicate actual core flow has lowered. The mismatch between JP 9 and 10 indicates one of these jet pumps has faulted. JP 10 flow higher, JP 9 flow lower and the opposite JP loop total flow higher are all consistent with a displaced jet pump mixer. This is confirmed by the rise in Recirc Pump B flow, as the displaced mixer allows the riser pipe to discharge directly into the downcomer.**

Safety function of a jet pump is described in the TS 3.4.3 Bases. Jet pump structural integrity is required to ensure the core can be reflooded to 2/3 core height after the DBA LOCA.

A **Correct. The symptoms presented are consistent with a displaced JP mixer. JP safety function is lost as 2/3 core flooding cannot be assured with a failed mixer.**

B **Incorrect. For a loose JP mixer the JP flow of both JP on the riser is expected to lower. JP safety function is lost as 2/3 core flooding cannot be assured with a failed mixer.**

C **Incorrect. For a loose JP mixer the JP flow of both JP on the riser is expected to lower. JP safety function is lost with the structural failure of 1 JP.**

D **Incorrect. For a plugged JP nozzle indicated core flow is consistent with actual JP flow. Recirc Pump B flow would be lower due to the increased flow resistance of the plugged nozzle.**

10CFR55	**41.3**
Technical References	**ON-164-005** **TS 3.4.2 Bases**
Learning Objectives	**11502**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer mj / 05/14/14
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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	4
K/A		295024 EA2.06 High Drywell Pressure				Importance		4.1	
Statement		Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Suppression pool temperature							

QUESTION 53

Which one of the following identifies a set of initial conditions that could lead to Primary Containment pressure exceeding the design limit if a design-basis Loss of Coolant accident were to occur?

A. Drywell pressure at 0.6 psig
Suppression Chamber pressure at 1 psig

B. Suppression Pool temperature > 105 °F
HPCI full-flow test in progress

C. Both loops of Drywell spray inoperable

D. 2 of 3 required Drywell cooler fan pairs inoperable

Proposed Answer: **B**

Applicant References: **None**

Explanation: **The applicant is required to evaluate 4 postulated initial conditions to identify the initial condition that lies outside the assumptions of the DBA LOCA analysis such that the high Drywell pressure design limit could be exceeded.**

A Incorrect. Both primary containment compartment pressures are within the TS 3.6.1.4 LCO requirements. Although the Drywell is typically slightly positive relative to the Suppression Chamber, the only specific requirements on ΔP is < 1.5 psid DW-SC per TS 3.6.1.4 and > -0.5 psid DW-SC to prevent opening vacuum breakers.

B Correct. This Suppression Pool temperature, combined with continued testing that results in adding heat to the Suppression Pool, could result in exceeding Drywell high pressure design limits in the DBA LOCA due to being outside the initial conditions assumed in the containment pressure and pool heatup analyses. TS 3.6.2.1 Condition C requires immediate action to limit continued SP temperature increase and hourly action to monitor SP temperature to ensure the reactor operating limit of 110 °F is not exceeded.

C Incorrect. Drywell spray is the primary means for rapidly lowering DW pressure following events that result in high Drywell pressure. However, functionality of RHR for DW spray is not required by Technical Specifications or the TRM.

D Incorrect. The operability of 3 pairs of Drywell cooler fans is required by TS 3.6.3.2. The safety function of the DW cooler fans is for containment atmosphere mixing post-LOCA to dilute any hydrogen produced throughout the entire DW volume. Action within 1 hour is required if 2 of 3 pairs are inoperable, but the action is to verify the alternate hydrogen control function of containment nitrogen purge is available. The cooling function of the DW coolers is not required to be operable to ensure the DW pressure response post-LOCA is acceptable.

10CFR55: **41.9**

Technical References: **TS 3.6.2.1 Bases**

Learning Objectives: **13430**

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Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer mj / 03JUN14
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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A		295023 2.4.18 Refueling Accidents				Importance		3.3	
Statement		Knowledge of the specific bases for EOPs.							

QUESTION 54

Which one of the following describes why restarting Reactor Building HVAC, bypassing LOCA interlocks if necessary, is allowed by EO-000-104, Secondary Containment Control?

A. Maintain personnel access to Secondary Containment during post-accident conditions to operate equipment needed to reduce offsite radioactivity release

B. Maintain functionality of equipment located in Secondary Containment during events where the potential for radioactive release is low

C. Return Reactor Building Zone III normal ventilation to service during long-duration events to assist with Spent Fuel Pool cooling

D. Minimize spread of airborne contamination from the unit experiencing the accident to the unaffected unit

Proposed Answer: **B**

Applicant References: **None**

Explanation: **Restarting Reactor Building normal HVAC is allowed by EO-000-104 Step SC-3 under certain specific conditions. Restarting HVAC is important to re-establish building cooling, allowing personnel access and maintaining conditions in the RB within the environment qualifications of equipment important to safety located in the RB. Restoration of RB HVAC results in untreated releases from the secondary containment via the normal RB exhaust. Bypassing isolation logics and restoring normal ventilation is therefore not allowed when there is potential for radioactive release due to returning normal HVAC systems to service.**

A Incorrect. Maintaining personnel access to Secondary Containment is important, but restoring normal HVAC for the purpose of entering Secondary Containment would result in higher release rates and is not authorized by EO-000-104 Step SC-3.

B Correct. Maintaining RB conditions within the EQ limits for equipment located in Secondary Containment, when no significant release is expected, is the reason to re-establish normal RB HVAC.

C Incorrect. ON-135-001 does include instructions for preparing RB HVAC systems for operation during a complete loss of Spent Fuel Pool cooling. These instructions presume the isolation of Zone 3.

D Incorrect. Isolation of either units normal RB HVAC (Zone I or II) will also initiate a Zone III (common zone) isolation. Zones I and II are separate and isolated from communication in the normal lineup. Isolation of Zone III isolates the common recirculation space from the non-affected unit preventing the spread of airborne radioactivity to the unaffected unit.

10CFR55: **41.10**

Technical References: **EO-000-104 Step SC-3**

Learning Objectives: **14613**

Question Source: **New**

SUSQUEHANNA STEAM ELECTRIC STATION
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Previous NRC Exam **No**

Comments

Operations Reviewer mj / 05/14/14
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Exam	RO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A	295003 2.4.49 Partial or Complete Loss of A.C. Power					Importance		4.6	
Statement	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.								

QUESTION 55

Units 1 and 2 are operating at rated power.

ESS Transformer T-111 experiences a transformer lockout.

ESS Bus 2C is de-energized by the transformer lockout.

Breaker 2A203-01, XFMR 111 TO BUS 2C, remains closed.

Which one of the following identifies the immediate action required in response to this condition?

A. Open breaker 2A203-01, ONLY

B. Open breaker 2A203-01
Close breaker 2A203-08, XFMR 211 TO BUS 2C

C. Turn XFMR 211 TO BUS 2C synchroscope on
Close breaker 2A203-08, XFMR 211 TO BUS 2C

D. Place Diesel Generator C governor control to isochronous
Depress DG C start pushbutton

Proposed Answer: A

Applicant References: None

Explanation: With the electric plant in the normal alignment ESS Transformer T-111 is the normal feeder to ESS Bus 2C. On a transformer lockout the transformer feeder breaker from other Startup Bus opens, and all downstream feeders from the transformer open. For ESS Bus 2C this is 2A203-01. This breaker remaining closed represents a failure of a protective action to occur automatically. Per OP-AD-001 Section 6.4 the operator shall manually initiate the protective feature should it fail to occur automatically. In this case that is to open 2A203-01. Once 2A203-01 opens the bus transfer scheme to its alternate supply should occur automatically. Energization of a dead ESS 4KV bus, if required, is performed per ON-004-002.

A Correct. 2A203-01 should have opened automatically on the transformer lockout. This is the only action required to be performed immediately in response to the transformer lockout.

B Incorrect. Immediate action to close breaker 2A203-08 is not allowed per OI-AD-006 Step 4.3.15.a.

C Incorrect. Operation of the ESS Bus 2C synchroscope is not required to respond to the situation. Procedural direction for turning on synchroscope must be followed per ON-004-002.

D **Incorrect. Starting the DG will not re-energize the bus. The DG does not have a start signal, as the DG start logic still sees the 2A203-01 breaker closed. The DG start logic does not include a direct start signal on bus undervoltage, only normal and alternate feeder breakers open.**

10CFR55	**41.10**
Technical References	**AR-015-E01** **OP-AD-001 Step 6.4** **ON-104-203 Section 5.0** **OI-AD-006 Step 4.3.15b**
Learning Objectives	**10121**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

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Exam	RO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	4
K/A	295019 2.4.9 Partial or Complete Loss of Instrument Air					Importance	3.8		
Statement	Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.								

QUESTION 56

Unit 1 is shutting down for a planned outage and is in Mode 3.

Feedwater pumps are shutdown and isolated.

Operators are preparing to establish Condensate long-path recirculation flow with the LV-10641 Startup Level Control valve.

A small leak develops in the Drywell. A reactor scram occurs on high Drywell pressure.

A loss of Instrument Air occurs.

Reactor level is -5", down slow.

Which one of the following describes the immediate availability of Condensate to restore reactor level, and the reason why?

	Condensate Availability	Reason
A.	NOT available	Startup level control valve LV-10641 cannot be opened
B.	NOT available	Condensate Filtration System inlet and outlet valves fail closed
C.	Available	Startup level control bypass valve HV-10640 valve remains functional
D.	Available	Startup level control valve (LV-10641) fails as-is

Proposed Answer **A**

Applicant References **None**

Explanation

A loss of instrument air has a number of effects on the Condensate system. The condensate pump min flow valves fail open, diverting Condensate back to the hotwell, minimizing the injection capability of the system at higher pressures. At lower pressures the system may be capable of some injection to the reactor. Condensate pumps are not directly affected by the loss of air, as pump cooling is maintained.

The startup level control valve LV-10641 fails as-is on a loss of air. The 10641 valve is closed in preparation for the long-path recirculation alignment

A Correct. A flow path to align Condensate Pumps to inject to the reactor cannot be established in the Control Room. The LV-10641 was closed at the time of the loss of air and fails as-is.

B Incorrect. The CFS inlet and outlet valves fail as-is on a loss of I/A.

C Incorrect. Condensate injection would not be available as the 10640 fails closed. This distractor is consistent with the misunderstanding of the method of operation of the 10640, due to the HV designation typically used for MOVs and the lack of automatic valve control.

D Incorrect. The LV-10641 fails as-is, closed in preparation to establish long-path recirc flow.

10CFR55	41.4	
Technical References	ON-118-001	
Learning Objectives	11155 a	
Question Source	Modified Bank	Vision LOC_BASIC S-300000-RBO-10-Q02. Revised stem conditions and correct answer. Additional changes for revision of ON-118-001.
Previous NRC Exam	No	
Comments	Reference CR 2014-16675 for changes in LV-10641 operation.	

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Exam	RO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		**295025 EK1.03 High Reactor Pressure**				Importance		3.6	
Statement		**Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : Decay heat generation**							

QUESTION 57

Unit 1 is starting up at 14 percent power.

Reactor Feedpump B has been placed in Flow Control Mode, with valve control for all 3 RFPs in MANUAL, due to a suspected software error in ICS.

The reactor is manually scrammed due to trip of both Recirc Pumps, per ON-100-101, Scram, Scram-Imminent.

In the scram report, one minute after scram, reactor pressure is reported as 790 psig, down slow, with MSIVs open.

Which one of the following characterizes the reactor pressure response, and the prompt operator action required in response to these conditions?

	Reactor pressure response	Operator action
A.	Lower than expected	Close MSIVs due to PCIS malfunction
B.	Lower than expected	Close bypass valves with the manual jack
C.	Expected	Close MSIVs to prevent violating cooldown rate
D.	Expected	Manually align Feedwater in startup level control to prevent uncontrolled injection

Proposed Answer **D**

Applicant References **None**

Explanation

Following a reactor scram from low-power (approximately 14 percent) at beginning of cycle, core decay heat is at a minimum and reactor pressure following a scram will be slow to recover. Prompt action with reactor pressure at 790 psig and going down will be required to ensure FW realigns to the startup level control alignment.

Operation at 11-15% RTP with a RFP in FCM is allowed by GO-100-102.

Low reactor pressure following a low-power scram is expected. The plant-reference simulator shows reactor pressure at 750 psig and lowering 150 seconds after a manual scram with no recirc pumps running.

A Incorrect. Conditions for an automatic closure of the MSIVs were not met as the reactor was manually scrammed from power. ON-100-101 directs placing the Mode switch to SHUTDOWN to scram the reactor, bypassing the MSIV closure on low pressure.

B Incorrect. No reason to expect bypass valve malfunction is provided in the stem. The manual jack would be ineffective in closing failed open bypass valves.

C Incorrect. Prompt action to close MSIVs is not procedurally directed for these conditions. Additional actions to close MSL drains, secure a RFP, and realign aux steam to secure normal steam loads should be attempted first.

D Correct. With the initial low reactor pressure this low and trending down, action to realign FW to startup level control is appropriate. EO-100-102 Step RC/P-1 will provide direction for this action once EO-100-102 is entered.

10CFR55	41.5
Technical References	OP-145-001 Att A EO-000-102 Step RC/P-1 GO-100-102 ON-100-101
Learning Objectives	16095
Question Source	New
Previous NRC Exam	No
Comments	

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Exam	RO	Tier	1	Group	1	Cognitive Level	Low	Level of Difficulty	2
K/A		295006 AK3.06 SCRAM				Importance		3.2	
Statement		Knowledge of the reasons for the following responses as they apply to SCRAM : Recirculation pump speed reduction							

QUESTION 58

Which one of the following identifies the reason the Recirc Pumps runback to LIM1 following a reactor scram at power?

A. To reduce power in the upper portion of the core by increasing voiding

B. To minimize reactor level shrink during the scram transient

C. To provide a redundant method of core power reduction

D. To maintain Recirc Pump NPSH

Proposed Answer: **D**

Applicant References: **None**

Explanation: **The Recirc Pumps runback to LIM1 on a reactor scram on a +13" reactor level signal or low FW flow. The purpose of this runback on low level is to maintain recirc pump NPSH due to the loss of static head to the recirc pump suctions. The purpose of the low FW flow runback is to maintain recirc pump NPSH with higher temperature water in the downcomer.**

A **Incorrect. This is the reason for the EOC-RPT function, which trips the recirc pumps to lower power to improve MCPR margin during the turbine trip transient.**

B **Incorrect. Reducing recirc pump speed has the effect of raising downcomer levels, but this is not done to affect reactor level during the scram transient.**

C **Incorrect. This is the reason for the ATWS-RPT function, which trips the recirc pumps to off on lower reactor levels which could be indicative of an ATWS condition.**

D **Correct. The basis for the LIM1 runback on low reactor level is to limit recirc pump speed to maintain NPSH.**

10CFR55: **41.5**

Technical References: **AR-102-C01**
TM-OP-064E

Learning Objectives: **16026**

Question Source: **New**

Previous NRC Exam: **No**

Comments

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RECIRC A FLOW LIMIT RUNBACK (C01)

SETPOINT: Not Applicable

ICS RECIRC/A MG Drive Motor Breaker Closed AND LIM #1 RR1A08-1/LIM #2 RR1A08-3

1. PROBABLE CAUSE:

1.1 Recirc runback 48% caused by following:
  1.1.1 Low reactor water level 30" WITH Feedwater Heater #1 OR #2 Hi Hi level.
  1.1.2 Condensate pump trip.
  1.1.3 Low feedwater pump flow ≤ 16.4% (.9Mlbm/hr).
  1.1.4 Circ water pump tripped condition present and Condenser Pressure equal to or greater than 6.0" HgA.
  1.1.5 Manual Flow Reduction to the 48% Speed Limiter.

1.2 Recirc runback 30% caused by any of following:
  1.2.1 Total feedwater flow ≤ 16.4% (2.7Mlbm/hr) for 15 seconds or
  1.2.2 RECIRC PUMP A DSCH HV-143-F031A not full open.
  1.2.3 Reactor vessel water low level 3.
  1.2.4 Manual Flow Reduction to the 30% Speed Limiter.

2. OPERATOR ACTION:

☐ 2.1 **Ensure** Automatic Actions.
☐ 2.2 **Perform** ON-164-002 Loss of Reactor Recirculation Flow.

3. AUTOMATIC ACTION:

☐ Recirc runback to applicable limit.

4. REFERENCE:

4.1 E-323 Sh 29
4.2 E-129 Sh 1, 17
4.3 E-151 Sh 2
4.4 M1-B31-275(13)
4.5 IOM 305
4.6 E-16 Sh 17
4.7 FD-1304 sh 1, 2; FF62208 sh 33, 34
4.8 FD-1305 sh 1, 2; FF62208 sh 35, 36

Exam	RO	Tier	1	Group	2	Cognitive Level	High	Level of Difficulty	4
K/A		295002 AK1.04 Loss of Main Condenser Vacuum				Importance		3.0	
Statement		Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM : Increased Offgas flow							

QUESTION 59

Unit 1 is operating at rated power.

Annunciator STEAM SEAL EVAP HI-LO LEVEL (AR-119-B01) is received.

Seal Steam Evaporator level indicated on LI-10749, SSE LEVEL, indicates -2.5 inches, down fast.

Which one of the following identifies

(1) the appropriate action to take to clear the alarm?

(2) the action required if Seal Steam is lost and CANNOT be recovered?

	Action to clear alarm	Action if Seal Steam lost
A.	Close SSE BLOWDOWN ISO, HV-101761	Scram the reactor Close MSIVs
B.	Close SSE BLOWDOWN ISO, HV-101761	Perform Scram Imminent Actions Place second Offgas charcoal subtrain in-service
C.	Open SSE LEVEL BYPS, HV-10750	Scram the reactor Close MSIVs
D.	Open SSE LEVEL BYPS, HV-10750	Perform Scram Imminent Actions Place second Offgas charcoal subtrain in-service

Proposed Answer C

Applicant References None

Explanation

A malfunction in the condensate supply to the Seal Steam Evaporator has occurred as indicated by the SSE high-low level and indicated SSE level at the low-level alarm setpoint and lowering. Makeup to the SSE is required to maintain seal steam header pressure and prevent a loss of Main Condenser vacuum. The appropriate action to attempt to clear the alarm is to open the bypass around the normal SSE level control valve, HV-10750. If seal steam is completely lost, air intrusion past the turbine seals will result in a total loss of condenser vacuum. The reactor must be scrammed and the MSIVs must be closed in anticipation of the turbine trip and automatic MSIV isolations that occur on low condenser vacuum, combined with the possibility of seal damage due to excessive cold air flow across the hot labyrinth seals.

A Incorrect. The SSE blowdown isolation valve is the isolation valve to the continuous blowdown line to the Main Condenser. Closing this isolation valve will have a momentary effect on SSE level, but additional makeup will be required.

B Incorrect. The SSE blowdown isolation valve is the isolation valve to the continuous blowdown line to the Main Condenser. Closing this isolation valve will have a momentary effect on SSE level, but additional makeup will be required. Normally the SSE drains to the #2 FW heaters for improved thermal efficiency. While placing a 2nd charcoal train in-service is required for Offgas flow > 150 scfm, for a total loss of seal steam condenser vacuum cannot be maintained. A reactor scram and MSIV closure will occur.

C Correct. This will result in additional makeup to the SSE if condensate transfer is in service to clear the SSE low-level alarm. The reactor must be scrammed and the MSIVs must be closed in anticipation of the turbine trip and automatic MSIV isolations that occur on low condenser vacuum when seal steam is totally lost.

D While placing a 2nd charcoal train in-service is required for Offgas flow > 150 scfm, for a total loss of seal steam condenser vacuum cannot be maintained. A reactor scram and MSIV closure will occur.

10CFR55	41.5
Technical References	AR-119-C02 ON-143-001 Step 3.7.4
Learning Objectives	10944 g
Question Source	New
Previous NRC Exam	No
Comments	

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Exam	RO	Tier	1	Group	2	Cognitive Level	High	Level of Difficulty	3
K/A	295017 AK2.14 High Off-Site Release Rate					Importance	4.0		
Statement	Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: PCIS/NSSSS								

QUESTION 60

Units 1 and 2 are in Mode 1.

A loaded Dry Fuel transfer cask is being moved to its trailer in the Central Railroad Bay.

The Central Railroad Bay is aligned to Zone 3 HVAC.

The cask drops, resulting in significant fuel damage and a breach of the cask confinement boundary.

Reactor Building HVAC exhaust duct radiation monitors indicate as follows (mR/hr):

	Refuel Floor High	Refuel Floor Wall	Railroad Access Shaft
Channel A	4	6	15
Channel B	3	9	Downscale

Which one of the following identifies how offsite releases will be minimized in this condition?

	Zone III	Reactor Building Recirc Fans	Standby Gas Treatment
A.	Automatically isolates	Both auto-start	Both auto-start
B.	Automatically isolates	Fan A auto-starts	Train A auto-starts
C.	Must be manually isolated	Fan A auto-starts	Train A auto-starts
D.	Must be manually isolated	Must be manually started	Must be manually started

Proposed Answer B

Applicant References None

Explanation

A fuel handling accident in the Zone III space of secondary containment has occurred. A ventilation exhaust process rad monitor has tripped. This results in an isolation signal to Zone III and a start signal to the A train of SBGT and the A RB Recirc Fan. Actuation of either channel of Zone III isolation logic results in a full isolation of the Zone (1 of 2 dampers in-series). The B RR Access Shaft rad monitor appears to have failed, perhaps due to the accident, in the downscale conditions. This results in a DOWNSCALE/INOP alarm, but no INOP trip. No Channel B rad monitor is in the tripped condition.

A Incorrect. Only the Division 1 components will auto-start due to the A channel exceeding the TRIP setpoint. The downscale does not generate any auto-start signals.

B Correct. Zone III isolates and the A SBGT and RB Recirc Fan start.

C Incorrect. Manual isolation of Zone III is not required.

D Incorrect. One train of SBGT and a RB Recirc Fan auto-start, which is sufficient to assure the safety function of minimizing release from the accident. Zone III automatically isolates.

10CFR55	41.9
Technical References	AR-016-F12,H12
Learning Objectives	10879 e
Question Source	New
Previous NRC Exam	No
Comments	

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Exam	RO	Tier	1	Group	2	Cognitive Level	High	Level of Difficulty	4
K/A	500000 EK3.07 High Containment Hydrogen Concentration					Importance	3.1		
Statement	Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Operation of drywell vent								

QUESTION 61

Refer to the table below when answering this question.

Unit 1 experienced a fuel-damaging severe accident.

Adequate core cooling was lost and could not be re-established with both loops of RHR aligned for LPCI.

Current Containment combustible gas concentrations are as follows:

	Hydrogen	Oxygen
Drywell	8 percent	4 percent
Suppression Chamber	2 percent	5 percent

Which one of the following describes the combustible gas control strategy for these conditions?

A. Vent the Drywell, to remove combustible gas from the Containment airspace to prevent a hydrogen deflagration

B. Vent the Drywell, to maintain Containment pressure as low as possible in the event of a hydrogen detonation

C. Spray the Containment, to cool non-condensibles and scrub fission products out of the Containment atmosphere before release

D. Maximize Containment Recombiner operation, to reduce combustible gas concentrations

TABLE 6
COMBUSTIBLE LIMITS

DW OR SUPP CHMBR	H_2	**6%**
AND		
DW OR SUPP CHMBR	O_2	**5%**

Proposed Answer A

Applicant References None

Explanation

Combustible gas concentrations have exceeded the limits of EO-100-103 Table 6. EO-103 actions require recombiners be secured (Step PC/G-4) and EP-DS-001 entered (Step PC/G-7). The strategies available in EP-DS-001 for combustible gas control include primary containment venting in addition to recombiner operation and containment spray.

A Correct. Venting the Drywell is the preferred combustible gas control strategy. Venting the DW is preferred to venting the Suppression Chamber for these conditions, as introducing the high H2 concentrations in the DW to the SC would create a combustible mixture.

B Incorrect. Attempting to lower DW pressure in anticipation of a hydrogen deflagration is not a recognized method of H2 control.

C Incorrect. Containment spray is not available as all RHR pumps are required to attempt to restore adequate core cooling.

D Incorrect. Recombiner operation is precluded with a combustible mixture present in Containment.

10CFR55 41.5

Technical References

EO-100-103 Step PC/G-4
EP-DS-001 Section 1
EP-DS-004 Att A

Learning Objectives 12098

Question Source New

Previous NRC Exam No

Comments

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Exam	RO	Tier	1	Group	2	Cognitive Level	High	Level of Difficulty	4
K/A		295008 AA1.07 High Reactor Water Level				Importance		3.4	
Statement		Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL : Main turbine							

QUESTION 62

Unit 1 is operating at rated power.

The 1C004 instrument rack experiences a leak on the common Narrow Range level variable leg. All Narrow Range level indications on the 1C004 rack begin drifting lower.

Feedwater Level Control marks Narrow Range Level Channel B as DEVIANT.

Which one of the following identifies the effect on reactor level, and the operator action to be taken due to the level instrument malfunction?

	Reactor level effect	Operator action
A.	No effect	Place 1C004 in Maintenance Bypass
B.	Reactor level lowers as FWLC reduces FW flow	Insert a manual scram
C.	Reactor level rises as FWLC increases FW flow	Select Narrow Range A or C for FWLC
D.	Reactor level rises as FWLC increases FW flow	Scram the reactor Trip the Main Turbine and all Reactor Feedwater Pumps

Proposed Answer D

Applicant References None

Explanation A variable leg leak on the 1C004 instrument panel results in slowly lowering reactor level indications on the N004A and C inputs to ICS and the N024A and B inputs to RPS, among others. With ICS marking the unaffected NR input N004B as DEVIANT, the ICS level selection logic is taking the A and C inputs as indicated reactor level. As these indications are drifting lower, ICS begins raising FW flow to attempt to raise level. With no feedback due to the instrument drift, actual reactor level continues to rise and will eventually reach +54". With the NR A and C indicating low, no turbine trip signal will be generated.

A Incorrect. Actual level will rise. Placing 1C004 in Maintenance Bypass would be an appropriate response to the malfunction.

B Incorrect. Actual level will rise. The action of inserting a manual scram is consistent with the assumption that 1 division of RPS has failed due to multiple level instrument failures.

C Incorrect. While actual reactor level would rise, selecting one of the failed instrument channels for FWLC is an inappropriate action. This distractor is consistent with failing to identify the instruments associated with the C004 rack.

D Correct. Actual level is rising, and the Main Turbine and RFPT trips at +54" are disabled with the failure of the NR A and C lower.

10CFR55	41.7
Technical References	ON-145-001 ON-145-004
Learning Objectives	10297 I
Question Source	New
Previous NRC Exam	No
Comments	

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Exam	RO	Tier	1	Group	2	Cognitive Level	High	Level of Difficulty	2
K/A		**295022 AA2.02 Loss of CRD Pumps**				Importance		3.3	
Statement		**Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : CRD system status**							

QUESTION 63

Unit 1 has experienced an ATWS.

Operators maximized CRD per EO-100-113 Sheet 2.

Subsequently, CRD Pump suction pressure lowered to 5" HgV for 4 seconds, then returned to normal.

Annunciator CRD PUMP SUCTION FILTER HI DIFF PRESS (AR-107-C01) was in alarm momentarily, but has now cleared.

Which one of the following identifies the required operator action with regards to the CRD pump suction filter to continue to attempt to drift control rods?

A. Bypass the CRD Pump suction filter, ONLY

B. Lower the output of the CRD flow controller
THEN
Bypass the CRD Pump suction filter

C. Bypass the CRD pump suction filter
THEN
Restart both CRD Pumps

D. Restart both CRD Pumps
Bypass the CRD pump suction filter ONLY if the alarm re-flashes

Proposed Answer **C**

Applicant References **None**

Explanation **With both CRD pumps running the CRD pump suction filter has clogged and resulted in a trip of both CRD pumps on low suction filter. ON-155-007 Section 3.6 provides instructions for bypassing the pump suction filter and restarting CRD Pumps if tripped. In this event, both CRD pumps tripped on low suction pressure for more than the 3-sec TD.**

A Incorrect. Both CRD Pumps have tripped. Bypassing the CRD pump suction filter alone is inadequate to attempt to drift control rods.

B Incorrect . Reducing the flow through the system would be an appropriate action if the CRD pumps were still running

C Correct. Both CRD pumps have tripped. The pump suction filter must be bypassed before the pumps can be restarted and kept running.

D Incorrect. The alarm cleared only due to the trip of both CRD pumps. The CRD pumps will continue to trip on low suction pressure if the suction filter is not bypassed.

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10CFR55	**41.6**	
Technical References	**AR-107-B01, C01** **ON-155-007 Section 3.6**	
Learning Objectives	**11444 m**	
Question Source	**Bank**	**LOR LXR AD045/15304/145**
Previous NRC Exam	**No**	
Comments		

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Exam	RO	Tier	1	Group	2	Cognitive Level	High	Level of Difficulty	2
K/A		295007 2.4.6 High Reactor Pressure				Importance		3.7	
Statement		Knowledge of EOP mitigation strategies.							

QUESTION 64

Unit 1 is operating at rated power.

HPCI is out of service for routine maintenance.

The reactor scrams from rated power due to a loss of EHC.

Which one of the following identifies the first method of manual pressure control capable of stabilizing reactor pressure below the scram setpoint per EO-000-102, RPV Control?

A. Main Turbine Bypass Valves, using the manual jack

B. Main Steam Line drains

C. Align RCIC for CST-to-CST operation

D. SRVs using an A-B-C sequence

Proposed Answer: D

Applicant References: None

Explanation: Following a loss of Main Turbine EHC the main condenser remains available. Of the methods listed and available, only SRVs have enough capacity to maintain reactor pressure below the RPS scram setpoint.

A Incorrect. The manual jack is unavailable due to the loss of EHC. This distractor represents application of a motor actuator to the bypass valves similar to that used for the Main Turbine turning gear.

B Incorrect. MSL drains remain available on a loss of EHC, and Main Condenser availability is maintained. However, drain capacity is limited and will result in reactor pressure rising above the scram setpoint and SRV cycling on the relief setpoint.

C Incorrect. Use of RCIC for pressure control is allowed, however the capacity of RCIC is limited and inadequate to prevent reactor pressure rising above the scram setpoint and SRV cycling on the relief setpoint.

D Correct. SRVs provide the initial RPV pressure relief on an abrupt loss of EHC, and subsequent manual use will be required to maintain pressure in a stable band below the scram setpoint until another system can be recovered or decay heat lowers to within the capability of available systems.

10CFR55: 41.5

Technical References: EO-000-102 Step RC/P-6

Learning Objectives: 14593

Question Source: New

Previous NRC Exam: No

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Exam	RO	Tier	1	Group	2	Cognitive Level	High	Level of Difficulty	4
K/A		295029 EA2.03 High Suppression Pool Water Level				Importance		3.4	
Statement		Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Suppression pool water level							

QUESTION 65

Unit 1 experienced a fuel-damaging severe accident.

EP-DS-002, RPV and Primary Containment Flooding, is being performed.

The TSC has requested a determination if Containment water level has reached 116 ft, to see if core submergence has been achieved, using ON-159-003, Primary Containment Water Level Anomaly.

Which one of the following describes how the determination of Containment water level is to be made?

A. Plot Drywell pressure on the Containment level versus Drywell pressure graph, ONLY

B. Ensure the Drywell has been vented to atmosphere
THEN
Plot Drywell pressure on the Containment level versus Drywell pressure graph

C. Calculate the Drywell to Suppression Chamber ΔP
THEN
Plot the ΔP on the Containment level versus ΔP graph

D. Ensure the Drywell has been vented to atmosphere
THEN
Calculate the Drywell to Suppression Chamber ΔP
THEN
Plot the ΔP on the Containment level versus ΔP graph

Proposed Answer: **B**

Applicant References: **None**

Explanation: **TAF is a Containment water level of 116'. With a maximum indicated Containment water level of 49' on installed instrumentation, Suppression Chamber and Drywell pressures must be used to determine actual level. A level of 116' is in the Drywell above the Drywell pressure tap. Therefore Drywell pressure can be used to directly determine the water level in Containment, if the Drywell is vented to atmosphere.**

- A **Incorrect. Without ensuring the DW is vented to atmosphere, using DW pressure to determine Containment water level could give a false high reading.**
- B **Correct. With the DW vented to atmosphere, the DW pressure directly correlates to Containment water level.**
- C **Incorrect. This is the method used to determine Containment water level when level is above 49' and below the Drywell pressure tap at 64'.**

D Incorrect. This is the method used to determine Containment water level when level is above 49' and below the Drywell pressure tap at 64'. Containment pressurized above atmosphere will affect both SC pressure and DW pressure readings equally.

10CFR55	41.9	
Technical References	ON-159-003 EP-DS-002, Step RF-16	
Learning Objectives	337 a	
Question Source	Modified Bank	2011 LOC23 NRC Exam Question 64. Stem conditions changed to result in a different correct answer, minor editorial and formatting changes.
Previous NRC Exam	Yes	
Comments		

Operations Reviewer mo / 03/04/14
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Exam	RO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	3
K/A		**2.1.37 Conduct of Operations**				Importance		**4.3**	
Statement		**Knowledge of procedures, guidelines, or limitations associated with reactivity management.**							

QUESTION 66

Which one of the following evolutions requires a Reactivity Manipulation Package with a Reactivity Maneuver Request in accordance with OP-AD-338?

A. Lowering Recirc Pump speed from 35 to 30 percent for shutting down a Recirc Pump for Single Loop Operation per OP-164-001

B. Adjustments to recirc flow to maintain rated power, as xenon builds in following a plant startup

C. Movement of partially withdrawn control rods for monthly surveillance testing performed as part of SO-156-001, Control Rod Exercising

D. Movement of control rods performed as part of control rod scram time testing in Mode 1 per SR-155-004, Scram Time Measurement of Control Rods

Proposed Answer: **D**

Applicant References: **None**

Explanation: **OP-AD-338 Section 6.3.2 provides a list of reactivity maneuvers that do not require a Reactivity Maneuver Package.**

A **Incorrect. Lowering recirc pump speed for recirc pump shutdown is specifically exempted in OP-AD-338 Step 6.3.2b(5).**

B **Incorrect. Changes in recirc flow to maintain a specified power level, in this case < rated power, are specifically exempted in OP-AD-338 Step 6.3.2b(2).**

C **Incorrect. Performance of the monthly control rod push-me/pull-me surveillance is covered by the exemption in OP-AD-338 Step 6.3.2a(3), as the single-notch control rod movements in the SO will not change power by 5 percent.**

D **Correct. Moving control rods for scram time testing per SR-1(2)55-004i s not included on the list of activities exempted from requiring a RMP. SR 1(2)55-004 Step 5.4 states that a RMR will provide the authorization to stroke the control rods for performance of the test.**

10CFR55: **41.10**

Technical References: **OP-AD-338 Section 6.3.2**
NDAP-QA-0338 Step 5.13
SR-1(2)55-004 Step 5.4

Learning Objectives: **14913**

Question Source: **Bank** **AD044/14913/1 (LXR Ops Initial Bank)**

Previous NRC Exam: **No**

Comments

Operations Reviewer mj / 06/03/14
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6.3.2 The following reactivity evolutions do not require a Reactivity Manipulation Package Coversheet (Form OP-AD-338-6) and/or RMR (Form OP-AD-338-1) being the below activities are controlled by other approved procedures:

a. Control rod manipulations for:

(1) Controlled shutdowns/unplanned power reductions made in accordance with the Shutdown Control Rod Sequence package (Controlled Shutdown/Unplanned Power Reduction, Form OP-AD-338-5 and Shutdown Sequence Sheets).

(2) Full insertion of control rods per Shift Supervision direction due to emergency/off-normal plant conditions.

(3) Satisfying functional unit and/or test procedures (e.g., SO, TP, OT, etc.) that move control rods without impacting reactor power (see Definition 5.13 in NDAP-QA-0338).

(a) Any procedure that moves control rods in Mode 5 requires a Prerequisite to ensure proper blade support.

(b) Any procedures that move control rods in Modes 2, 3, and 4.

(c) Any procedures that move control rods in Mode 1 require a Prerequisite from RE to evaluate the targeted control rod movement.

(d) All control rod movements shall be documented within the requesting procedure or on an attached control rod movement sheet (Form OP-AD-338-2).

NOTE:	Form OP-AD-338-3 is not required for the recirculation flow manipulations listed below.

b. Recirc flow manipulations for:

(1) Emergency/unplanned power reductions (documented in Form OP-AD-338-5)

(2) Adjustments to flow to maintain a specific power level.

Exam	RO	Tier	3	Group	N/A	Cognitive Level	High	Level of Difficulty	2
K/A	2.1.25 Conduct of Operations					Importance	3.9		
Statement	Ability to interpret reference materials, such as graphs, curves, tables, etc.								

QUESTION 67

Refer to the figure on the following page when answering this question.

Unit 1 has experienced a large-break LOCA.

ECCS availability is limited. Only the following systems are injecting, and at the indicated flow rates:

Core Spray Pump 1B	3200 gpm
Core Spray Pump 1C	3500 gpm
RHR Pump 1C	8100 gpm

Compensated Fuel Zone Level indication is NOT available.

Reactor pressure is 200 psig.

Which one of the following correctly identifies the lowest non-compensated Fuel Zone level indication that provides adequate core cooling under these conditions?

A. -161"

B. -180"

C. -205"

D. -225"

Fuel Zone Indicated (INCHES OF WATER)	RPV Pressure (psig)											
	0	100	200	300	400	500	600	700	800	900	1000	1100
-110	-110	-93	-83	-76	-70	-64	-58	-53	-48	-43	-37	-32
	-120	-104	-95	-88	-82	-76	-71	-66	-61	-56	-51	-46
	-130	-114	-106	-99	-93	-88	-83	-79	-74	-69	-64	-60
	-140	-125	-117	-111	-105	-101	-96	-91	-87	-83	-78	-74
-150	-150	-136	-128	-122	-117	-113	-108	-104	-100	-96	-92	-88
TAF	-160	-147	-139	-134	-129	-125	-120	-117	-113	-109	-106	-102
	-170	-157	-150	-145	-141	-137	-133	-130	-126	-123	-119	-116
	-180	-168	-161	-157	-153	-149	-145	-142	-139	-136	-133	-130
	-190	-179	-173	-168	-164	-161	-158	-155	-152	-150	-147	-144
-200	-200	-189	-184	-180	-176	-173	-170	-168	-165	-163	-160	-158
	-210	-200	-195	-191	-188	-186	-183	-181	-178	-176	-174	-172
	-220	-211	-206	-203	-200	-198	-195	-193	-191	-190	-188	-186
	-230	-222	-217	-214	-212	-210	-208	-206	-204	-203	-201	-200
	-240	-232	-228	-226	-224	-222	-220	-219	-218	-217	-215	-215
-250	-250	-243	-239	-237	-235	-234	-233	-232	-231	-230	-229	-229
	-260	-254	-251	-249	-247	-246	-245	-244	-244	-243	-243	-243
	-270	-265	-262	-260	-259	-258	-257	-257	-257	-257	-257	-257
	-280	-275	-273	-272	-271	-271	-270	-270	-270	-270	-270	-271
	-290	-284	-284	-283	-283	-282	-283	-283	-283	-284	-286	-285
-300	-300	-297	-295	-295	-295	-295	-295	-295	-296	-297	-297	-299
	-310	-307	-306	-306	-306	-307	-307	-308	-309	-310	-311	-313

Proposed Answer	B
Applicant References	None
Explanation	For the given set of conditions the only available means of adequate core cooling is by submergence. While the combination of Core Spray flow is > 6350 gpm, spray cooling requires that flow from a single Core Spray loop to ensure the spray flow is effective at cooling the uncovered portion of the core by direct spray impingement. Wide Range cannot be used to determine if adequate core cooling is satisfied as the indicatin provided is unstable and Fuel Zone is trending down, expected with degraded ECCS systems during a LOCA. With the compensated FZ indication not avaialble, to determine reactor level the nomograph of indicated Fuel Zone level to actual reactor level provided in Att D of ON-145-004 must be used. A Incorrect. Fuel Zone level of -161" does assure adequate core cooling, but raising level that high is not required to establish adequate core cooling. B Correct. For a reactor pressure of 200 psig Att D of ON-145-004 shows that actual reactor water is at TAF forfs an indicated FZ level of -180". C Incorrect. An indicated FZ level of -205" is below TAF. This is also the MZIRWL for steam cooling with no injection, but application of MZIRWL is inappropriate under the specified conditions because of ECCS flow. D Incorrect. An indicated FZ level of -225" is below TAF. This level does correspond to the actual reactor level for spray cooling of -210", but application of spray cooling is inappropriate under the specified conditions because the total spray flow is split between two Core Spray loops.
10CFR55	41.10
Technical References	ON-145-004, Step 3.3 and Att D EO-000-102 Step RC/L-2, RC/L-18
Learning Objectives	1480
Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer mf / 05/15/14
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Exam	RO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	3
K/A		2.2.39 Equipment Control				Importance		3.9	
Statement		Knowledge of less than or equal to one hour Technical Specification action statements for systems.							

QUESTION 68

Unit 2 startup is in progress.

Reactor pressure is 800 psig.

The in-service CRD Pump trips. The standby pump cannot be started.

Cross-tie of Unit 1 CRD to supply Unit 2 CRD has been directed.

Which one of the following correctly describes the conditions for placing the Mode Switch to SHUTDOWN per Technical Specifications?

A. 20 minutes after determining any one CRD accumulator is inoperable

B. 20 minutes after determining greater than one CRD accumulator is inoperable
AND
Any inoperable accumulator is associated with a withdrawn control rod

C. Immediately upon determining any one CRD accumulator is inoperable
AND
The inoperable accumulator is associated with a withdrawn control rod

D. Immediately upon determining greater than one CRD accumulator is inoperable
AND
Any inoperable accumulators are associated with a withdrawn control rod

Proposed Answer: C

Applicant References: None

Explanation: **TS 3.1.5 applies. Condition C is entered on any HCU accumulator becoming inoperable due to low gas pressure with reactor pressure < 900 psig. The action to verify the accumulator is associated with a fully inserted control rod is required immediately upon recognizing a loss of CRD charging water header pressure. If the inoperable accumulator is for a withdrawn control rod, Required Action C.1 cannot be performed within the Required Action Time and entry into Condition D is required.**

- **A** Incorrect. The 20 minute allowance of Condition B does not apply. The distractor is plausible in that this may be a prudent action to take based on the Note 2 to Step 3.2 of ON-255-007, but it is not required by Tech Specs.
- **B** Incorrect. This is the correct action if the candidate were to incorrectly apply the requirements of TS 3.1.5 Conditions B and D as if reactor pressure were > 900 psig.
- **C** Correct. Condition C is entered on the first inoperable HCU accumulator, and Condition D requires placing the Mode Switch to SHUTDOWN immediately if the inoperable accumulator is associated with a withdrawn control rod.
- **D** Incorrect. Condition C only requires 1 HCU accumulator to be inoperable for entry.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

10CFR55	**41.10**	
Technical References	**ON-255-002 Step 3.2** **Unit 2 TS 3.1.5**	
Learning Objectives	**13430**	
Question Source	**Bank**	**TMOP055/12725/1 LXR OPS_INITIAL_BANK**
Previous NRC Exam	**No**	
Comments		

Operations Reviewer __mj / 05/15/14__ **Init / date**

Facility Representative _____/_______ **Init / date**

Exam	RO	Tier	3	Group	N/A	Cognitive Level	High	Level of Difficulty	4
K/A		2.2.15 Equipment Control				Importance		3.9	
Statement		Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.							

QUESTION 69

Use your provided references to answer this question.

Unit 1 is operating at rated power when SRV G spuriously opens.

Indications for SRV G solenoids are as follows:

Handswitch (1C601)	AMBER lit, RED extinguished
ADS A (1C601)	RED Extinguished
ADS B (1C601)	RED lit
Handswitch (1C628)	AMBER lit, RED extinguished
Handswitch (1C631)	AMBER extinguished, RED lit

Which one of the following identifies the fuses to pull to close the SRV?

A. F3B and F4B

B. F25B and F26B

C. F45 and F46

D. F3B and F4B
F45 and F46

Proposed Answer A

Applicant References M1-B21-129 Sht 5, 6 (redacted for power supply designation)

Explanation The indications provided are consistent with a spurious energization of the Division 2 ADS solenoid for SRV G, SV-14113G2.

A Correct. Fuses F3B and F4B supply power to the Division 2 ADS solenoid for SRV G, SV-14113G2 per M1-B21-129 Sht 5.

B Incorrect. Fuses F25B and F26B supply power to the Division 2 ADS solenoid indication for SRV G only, per M1-B21-129 Sht 6. Pulling these fuses would extinguish the lit indicators for SRV G, but would not close the SRV.

C Incorrect. Fuses F45 and F46 supply power to the normal relief operation solenoid for SRV G, SV-14113G3 per M1-B21-129 Sht 7. Pulling these fuses would have no effect with the Division 2 ADS solenoid energized for the SRV.

D Incorrect. Pulling fuses F45 and F46 is not required to close the SRV.

10CFR55 41.7

Technical References M1-B21-129 Sht 5, 6

Learning Objectives 13701

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer _mj / 05/15/14_
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Exam	RO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	3
K/A	2.3.13 Radiation Control					Importance	3.4		
Statement	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.								

QUESTION 70

Unit 1 is operating at 2 percent power.

Maintenance personnel have entered the Drywell to perform emergent repairs on elevation 738'.

The PCOM notes that reactor power is rising unexpectedly.

Reactor power continues to rise until it exceeds 3 percent.

Which of the following actions must the PCOM take per NDAP-QA-0309, Primary Containment Access and Control?

A. Manually insert control rods to maintain power < 3 percent

B. Immediately place the Mode Switch to SHUTDOWN

C. Immediately direct personnel to exit the Drywell

D. Immediately direct personnel to move down to Drywell elevation 704'

Proposed Answer: **B**

Applicant References: **None**

Explanation: **NDAP-QA-0309 Section 6.5 provides guidance for control of reactor power during Drywell entries with the reactor operating. The primary purpose of the procedure is to prevent a significant rise in Drywell radiation levels.**

- A **Incorrect. While control rod insertion to reduce power is allowed up to 3 percent power, in this situation the unexplained nature of the power excursion takes precedence and a reactor scram is required to prevent unexpected increases in Drywell radiation levels.**
- B **Correct. NDAP-QA-0309 requires the PCO stationed at the reactor controls to initiate a reactor scram by placing the Mode Switch to SHUTDOWN on any unexpected power increase.**
- C **Incorrect. Immediately directing personnel to exit the Drywell would be appropriate, but is insufficient to prevent unexpected increases in Drywell radiation levels.**
- D **Incorrect. Immediately directing personnel to lower elevations of the Drywell may be appropriate, but is insufficient to prevent unexpected increases in Drywell radiation levels.**

10CFR55: **41.12**

Technical References: **NDAP-QA-0309**

Learning Objectives: **15314**

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Question Source	Bank	LOC23 NRC (originally SRO question, but stem conditions and answer unchanged, so designated as bank question)
Previous NRC Exam	Yes	LOC23
Comments		

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Exam	RO	Tier	3	Group	N/A	Cognitive Level	High	Level of Difficulty	2
K/A		2.3.4 Radiation Control				Importance		3.2	
Statement		Knowledge of radiation exposure limits under normal or emergency conditions.							

QUESTION 71

An Alert has been declared due to radioactivity release rates.

The release is still in progress, but release rates have stabilized.

All Emergency Response facilities have been activated.

Which one of the following identifies, in accordance with EP-PS-100:

1) the maximum Emergency Exposure Extension that can be authorized to protect plant equipment to terminate the release?

2) whose approval, in addition to the Radiation Protection Coordinator, is required?

A. 10 Rem
Shift Manager

B. 10 Rem
Emergency Director

C. 25 Rem
Shift Manager

D. 25 Rem
Emergency Director

Proposed Answer: **B**

Applicant References: **None**

Explanation:

With the declaration of an Alert and release rates stable, no immediate threat is postulated to large populations and no actions for life-saving are required. The maximum Emergency Exposure Extension allowed by EP-PS-001 Att MM is 10 Rem.

EP-PS-001 Att MM requires approval from the Radiation Protection Coordinator (RPC) and either the Emergency Director or Recovery Manager. The TSC has been activated and therefore the Shift Manager has turned over the Emergency Director function to his relief. With turnover of the ED function the Shift Manager can no longer approve Emergency Exposure Extensions.

A **Incorrect. While this is the correct dose extension, the Shift Manager can no longer authorize the extension.**

B **Correct. This is the correct dose extension, and the ED approves dose extensions for on-site personnel.**

C **Incorrect. This dose extension is not warranted under these conditions; this is the limit for life-saving actions or protection of large populations. The Shift Manager can no longer authorize the extension.**

D Incorrect. This dose extension is not warranted under these conditions. The ED may authorizes extensions, but not to this dose level.

10CFR55	41.12
Technical References	EP-PS-001 Att MM
Learning Objectives	15106
Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer mj / 05/15/14
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PPL EMERGENCY PERSONNEL DOSE ASSESSMENT AND PROTECTIVE ACTION GUIDE

NOTE

EMERGENCY EXPOSURE EXTENSION REQUEST and POTASSIUM IODIDE TRACKING FORMS are in EP-PS-001 procedure.

Exam	RO	Tier	3	Group	N/A	Cognitive Level	High	Level of Difficulty	3
K/A		2.4.47 Emergency Procedures/Plan				Importance		4.2	
Statement		Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.							

QUESTION 72

Additional information to answer this question is provided on the next page.

Unit 2 experienced an ATWS. Initial ATWS power was 10 percent.

SLC is injecting.

Initial SLC Tank level was 1950 gal.

The STA reports SLC has injected 925 gal.

RPV water level is -90", down slow.

RPV pressure is being maintained with SRVs at 900 psig.

Suppression Pool temperature is 165 °F, steady.

Which one of the following identifies the level and pressure control strategy allowed by EO-200-113 in these conditions?

A. Raise reactor level to the normal band
Lower reactor pressure to begin a cooldown

B. Raise reactor level to the normal band
Maintain reactor pressure in the current band

C. Maintain reactor level in the ATWS band
Lower reactor pressure to begin a cooldown

D. Maintain reactor level in the ATWS band
Maintain reactor pressure in the current band

HEAT CAPACITY TEMPERATURE LIMIT

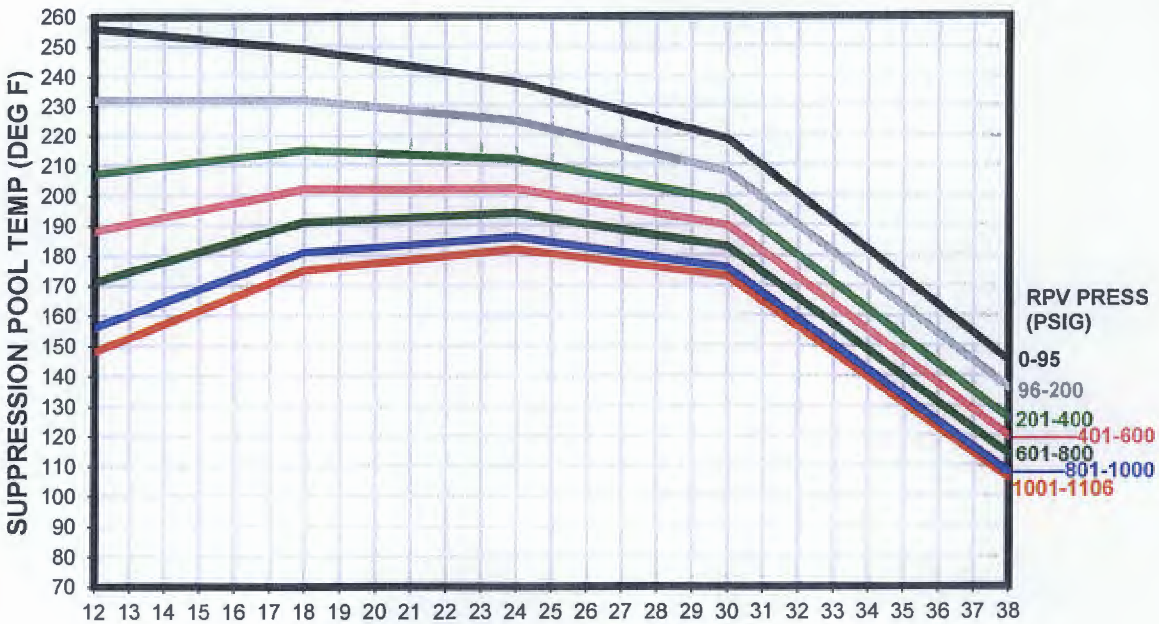

TABLE 19
HSBW INJECTED

INITIAL TANK VOLUME	FINAL TANK VOLUME
2000	1150
1900	1060
1800	975
1700	891
1600	806
1500	722
1400	637

Proposed Answer: B

Applicant References: None

Explanation: EO-200-113 Table 19 shows the Hot Shutdown Boron Weight for an initial SLC Tank level of 1950 gal is 1060 gal. Current tank level, calculated from the specified amount of boron solution injected, is 1025 gal, so the HSBW has been injected and reactor level is directed to be raised to the normal band by EO-200-113 Step LQ/L-16. A change in the pressure control band is not allowed by EO-200-113 at this time as the Cold Shutdown Boron Weight has not yet been injected (step LQ/P-8).

A Incorrect. Raising level is directed, but initiating a cooldown is not allowed by steps LQ/P-6 which requires pressure be stabilized. The pressure band specified is plausible in that it is the reactor pressure that would allow injection from condensate and does not require exceeding the 100 °F/hr cooldown rate. The pressure band is not allowed by EO-200-113 step LQ/P-4 as reactor level is being maintained with the available injection systems and violation of HCTL is not imminent.

B Correct. Raising level is directed when the HSBW is injected. This is the correct pressure band until the CSBW is injected.

C Incorrect. This is the correct level band with HSBW not yet injected. The pressure band specified is plausible as noted for Distractor A.

D Incorrect. If HSBW had not yet been injected this would be the correct level and pressure band.

10CFR55: 41.10

Technical References: EO-000-113
EO-000-103

Learning Objectives: 14594

Question Source: Modified Bank PP002/14594/097 LXR OP002_REQUAL_BANK. Changed stem conditions so HSBW had been injected, changing correct answer.

Previous NRC Exam: No Click here to enter text.

Comments: 2012 LOR Biennial

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Exam	RO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	3
K/A		2.4.22 Emergency Procedures/Plan				Importance		3.6	
Statement		Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.							

QUESTION 73

Unit 2 experienced an ATWS. Initial ATWS power was 100 percent.

Subsequently offsite power is lost. MSIVs close due to the loss of power.

RHR Suppression Pool cooling is maximized.

Rapid Depressurization is now required due to low reactor level.

Which one of the following identifies how RHR is to be operated for the Rapid Depressurization, and why?

	Action	Basis
A.	Continue RHR operation in Suppression Pool cooling	Maintain Suppression Pool temperature below the design limit
B.	Realign one division of RHR for LPCI	Re-establish adequate core cooling and maintain Suppression Pool temperature below the design limit
C.	Realign both divisions of RHR for LPCI and prevent injection	Allow manual control of LPCI flow to re-establish adequate core cooling
D.	Realign both divisions of RHR for LPCI	Maximize LPCI injection to re-establish adequate core cooling

Proposed Answer **C**

Applicant References **None**

Explanation **The isolated ATWS has resulted in Suppression Pool temperatures exceeding the point where operation of both loops of RHR in SP cooling is required by EO-200-103 step SP/T-2. The only exception to the requirement to maximize SP cooling is if RHR pumps are continuously needed for adequate core cooling. In this case adequate core cooling has been lost, as a Rapid Depressurization due to low reactor level is required.**

EO-200-113 step LQ/L-18 requires that injection from RHR Pumps be stopped before commencing a Rapid Depressurization to prevent uncontrolled injection and a large power excursion. A LPCI initiation signal is present on RHR as reactor level is below the initiation setpoint. OP-149-001. OP-149-001 Step 2.8.4 requires that the RHR Pumps be overridden OFF to prevent injection, as the LPCI injection valves will automatically open during the RD when reactor pressure falls below 420 psig.

A Incorrect. Continued operation of both Divisions of RHR in SP Cooling is not allowed by EO-200-103 due to the loss of adequate core cooling. Failure to prevent injection will result in uncontrolled injection from RHR during the Rapid Depressurization.

B Incorrect. While one division of RHR may be sufficient to restore adequate core cooling, failure to prevent injection on either division will result in uncontrolled injection.

C Correct. Both divisions of RHR must first be realigned for LPCI per Section 2.10 of OP-149-004 to prevent inadvertent draining of the RHR loops, then injection must be prevented.

D Incorrect. While both divisions of RHR may be required for adequate core cooling, injection must be prevented before the RD is initiated to prevent a power excursion from occurring due to uncontrolled injection.

10CFR55	41.10	
Technical References	EO-000-103 Step SP/T-2 EO-000-113 Step LQ/L-18 OP-149-001 Section 2.8 OP-149-004 Section 2.10	
Learning Objectives	10766, 14621	
Question Source	Bank	INPO 29211
Previous NRC Exam	No	
Comments		

Operations Reviewer mj / 05/15/14
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Exam	RO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	3
K/A		2.3.11 Radiation Control				Importance		3.8	
Statement		Ability to control radiation releases.							

QUESTION 74

Unit 1 is operating at rated power when annunciator OFF-GAS HI RADIATION (AR-106-G03) is received.

The reading from Off Gas Pre Treatment Log Radiation Monitoring recorder (RR-D12-1R601) is determined to be valid and has just exceeded Lim1.

Which one of the following identifies the next action required due to exceeding Lim1?

A. Scram the reactor and close the MSIVs and MSL drains

B. Immediately reduce power to lower Offgas pretreatment activity to < 150,000 µCi/sec

C. Contact Chemistry to obtain an Offgas pretreatment sample

D. Verify the Offgas system is not bypassed immediately

Proposed Answer: **C**

Applicant References: **None**

Explanation: **Offgas Hi alarm and Offgas readings exceeding Lim1 require entry into ON-179-002. The AR for the Offgas Hi alarm directs checking the readings on the Offgas pretreat recorder and evaluating entry into the ON. ON-179-002 describes Lim1 as set 50 percent above nominal steady-state background levels. With Lim1 set at a relatively low level this facilitates compliance with TS 3.7.5 for Offgas activity by ensuring pretreat samples are obtained to determine the actual Offgas activity level.**

A Incorrect. With Offgas pretreat readings just 50 percent higher than nominal background readings, MSL radiation levels will not have risen to the hi-hi alarm setpoint. Closure of the MSIVs is premature at this time.

B Incorrect. With Offgas pretreat readings just 50 percent higher than nominal background readings, actual Offgas activity levels remain at a very small fraction (<1 percent typically) of the TS 3.7.5 LCO limit. Action to reduce power to maintain Offgas activity less than half of the TS 3.7.5 limit will not be required with pretreat rad levels just exceeding Lim1.

C Correct. ON-179-001 Step 4.6 describes this action in response to Offgas pretreat readings above Lim1. Obtaining an Offgas pretreatment grab sample will allow determination of compliance with TS 3.7.5 limits.

D Incorrect. This is the TRM 3.7.7 Required Action and Completion Time for no operable Offgas pretreatment log radiation monitor. The question stem specifically identifies the reading as valid.

10CFR55: **41.11**

Technical References:
AR-106-G03
ON-179-002
TS 3.7.5
TRM 3.7.7

Learning Objectives	**15318**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer MJ / 03JUN14
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Exam	RO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	2
K/A	2.1.28 Conduct of Operations					Importance	4.1		
Statement	Knowledge of the purpose and function of major system components and controls.								

QUESTION 75

For the following RWCU following controls and indications

Control/Indication	Markings
(1) HV-144-F102, RWCU SUCTION	BROWN-striped pushbuttons (OPEN and CLOSE)
(2) HV-144-F001, RWCU INLET IB ISO	GREEN-collared handswitch
(3) FI-G33-1R609, RWCU INLET FLOW	PURPLE-RED label

Which one of the following correctly identifies the meaning of the handswitch and label colors?

A. (1) Containment isolation valve
(2) Throttlable flow-control valve
(3) Post-accident monitoring instrumentation

B. (1) Throttlable flow-control valve
(2) Containment isolation valve
(3) Reactor vessel flow instrumentation

C. (1) Containment isolation valve
(2) Throttlable flow-control valve
(3) DC-powered instrumentation

D. (1) Throttlable flow-control valve
(2) Containment isolation valve
(3) Nuclear heat balance instrument

Proposed Answer: **D**

Applicant References: **None**

Explanation: **The RWCU F102 valve is a throttleable system flow control valve. The RWCU F001 valve is an AC-powered containment isolation valve. The RWCU inlet flow indicator is used in the reactor core heat balance.**

A Incorrect. The F102 is the throttable valve, F001 is the PCIV. PAM instrumentation is not specifically given a unique label color at SSES, but is plausible as a group of instrumentation that could be specially designated.

B Incorrect. The F102 valve is throttleable, but the green collar designates ALL PCIVs, not just DC-powered PCIVs.

C Incorrect. The F102 valve is not a PCIV and is not DC-powered. The F001 is a PCIV and is not throttlable. The RWCU flow instrument is not DC-powered.

D Correct. The F102 is a throttlable valve, the F001 is a PCIV, and the RWCU flow instrument is a heat balance input.

10CFR55: **41.7**

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical References	**TM-OP-077** **E-165 Sht 6, 8**
Learning Objectives	**1376**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

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Exam	SRO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		600000 Plant Fire On-Site				Importance		3.1	
Statement		The fire's extent of potential operational damage to plant equipment							

QUESTION 76

Unit 2 is operating at rated power.

A fire breaks out in the Unit 2 Remote Shutdown Panel.

All 3 SRVs operated from Remote Shutdown Panel open and cannot be closed.

The reactor is scrammed from rated power.

Which one of the following identifies an appropriate response to stabilize the unit under these conditions, per ON-013-001?

A. Allow Condensate to flood the reactor to the main steam lines
Align Division 2 RHR in Suppression Pool cooling for long-term decay heat removal

B. Isolate the HPCI steam supply
Allow Condensate to flood the reactor to the main steam lines
Align Division 1 RHR in Suppression Pool cooling for long-term decay heat removal

C. Prevent uncontrolled Condensate injection by tripping all Condensate Pumps
Maintain reactor level with RCIC, until it isolates, then Division 1 Core Spray

D. Prevent uncontrolled Condensate injection by tripping all Condensate Pumps
Maintain reactor level with HPCI, until it isolates, then Division 2 Core Spray

Proposed Answer: **D**

Applicant References: **None**

Explanation: **Unit 2 is experiencing a fire in its Remote Shutdown Panel. Multiple SRVs open and the reactor is scrammed from rated power. The bases for ON-013-001 identifies that the preferred injection systems to use if available systems cannot maintain reactor level during s stuck-open SRV event is Division 2 Core Spray. The bases for ON-013-001 state that EOPs, ONs, GOs and other plant procedures will be utilized for shutdown.**

A **Incorrect. ON-013-001 does not identify a strategy of RPV flooding to respond to a fire in the Unit 2 Reactor Building. EO-200-102 requires reactor level maintained within the nominal band unless all reactor level indication is lost. For this fire, there is no threat identified to Division 2 indication, so entry into EO-200-114 for RPV Flooding is not expected. This is a strategy for a total loss of decay heat removal from ON-249-001, but the plant design for a worst-case fire in any area is to establish safe shutdown with 1 division of ESF equipment. Division 2 RHR will be available in Shutdown Cooling, entry into ON-249-001 will not be required.**

B **Incorrect. This distractor adds the guidance to override HPCI per ON-013-001 Att D Step D.7&8 to the direction provided in Distractor A.**

C Incorrect. While the actions to prevent uncontrolled condensate injection per EO-200-102 Step RC/P-1 is appropriate given the rapid lowering of reactor pressure expected for this event, ON-013-001 prefers the use of Division 2 systems due to the potential effects of the fire in the RSDP room.

D Correct. Actions to prevent uncontrolled condensate injection per EO-200-102 Step RC/P-1 are appropriate given the rapid lowering of reactor pressure expected for this event. Although the EOP bases describes the normal means of preventing uncontrolled injection is aligning Feedwater for Startup Level Control, in this transient action to trip the Condensate pumps would be appropriate. ON-283-001 Step 3.2 for stuck-open SRV provides similar guidance. ON-013-001 prefers the use of Division 2 systems due to the potential effects of the fire in the RSDP room per Step D.3 of Att D.

10CFR55	43.5	This is an SRO-level question as the requirements of EO and ON procedures must be evaluated given the plant conditions, and available equipment, in order to select the appropriate mitigating procedures consistent with ON-013-001 requirements.
Technical References	ON-013-001 Section 5.0, Att D Step D.3 EO-200-102, Step RC/P-1 ON-283-001 Step 3.2	
Learning Objectives	15304	
Question Source	New	
Previous NRC Exam	No	
Comments		

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Exam	SRO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A	700000 AA2.01 Generator Voltage and Electric Grid Disturbances					Importance	3.6		
Statement	Operating point on the generator capability curve								

QUESTION 77

Refer to the figure on the following page when answering this question.

Unit 1 is operating at rated power with main generator operation as shown.

Transient grid conditions result in oscillations in generator reactive load.

Main generator reactive load begins to oscillate between 200 and 300 MVAR.

Annunciator GEN VOLT REG AUTO TO MAN SETPOINT UNBALANCED (AR-106-C09) is in alarm.

Annunciator GENERATOR FIELD OVERVOLTAGE (AR-106-A06) remains clear.

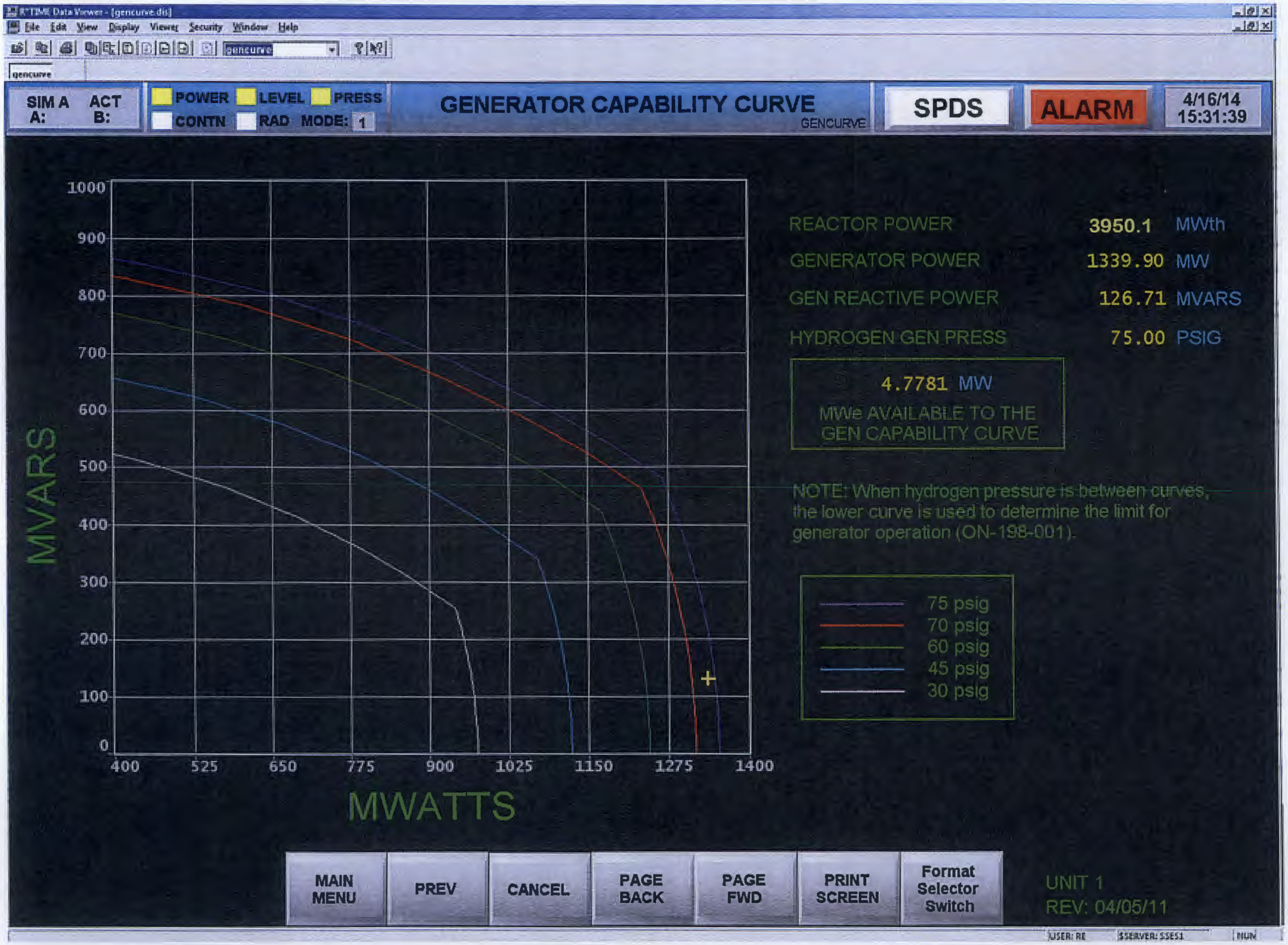
Which one of the following describes the appropriate actions to direct in response to the conditions represented by the process computer display?

A. Verify the Auto Voltage Regulator automatically maintains Generator Field current < 6000 amps
Adjust HC-10002, MAN VOLT REG ADJUST, as necessary to clear AR-106-C09

B. Immediately transfer to the Manual Voltage Regulator
Lower HC-10002, MAN VOLT REG ADJUST, until generator reactive load is < 150 MVAR

C. Reduce core power per the CRC instructions to lower generator load to restore positive margin to the capability curve
Perform GO-100-012, Power Operations for an unplanned power reduction

D. Immediately reduce core power per the CRC instructions to lower power by 5 percent
Perform GO-100-012, Power Operations for an unplanned power reduction


R*TIME Data Viewer - [gencurve.dis]
File Edit View Display Viewer Security Window Help
gencurve
SIM A A:
ACT B:
POWER
LEVEL
PRESS
CONTN
RAD
MODE: 1
GENERATOR CAPABILITY CURVE
GENCURVE
SPDS
ALARM
4/16/14 15:31:39
1000
900
800
700
600
500
400
300
200
100
0
MVARS
400
525
650
775
900
1025
1150
1275
1400
MWATTS
REACTOR POWER 3950.1 MWth
GENERATOR POWER 1339.90 MW
GEN REACTIVE POWER 126.71 MVARS
HYDROGEN GEN PRESS 75.00 PSIG
4.7781 MW
MWe AVAILABLE TO THE GEN CAPABILITY CURVE
NOTE: When hydrogen pressure is between curves, the lower curve is used to determine the limit for generator operation (ON-198-001).
75 psig
70 psig
60 psig
45 psig
30 psig
MAIN MENU
PREV
CANCEL
PAGE BACK
PAGE FWD
PRINT SCREEN
Format Selector Switch
UNIT 1
REV: 04/05/11
USER: RE
$SERVER: SSES1
NUM

Proposed Answer	**C**
Applicant References	**None**
Explanation	**ON-198-001 is the governing procedure for operation outside the generator capability curve with the Main Generator voltage regulator in AUTO. The initial conditions presented show operation just inside the limits of the capability curve. The transient results in sustained operation outside of the capability curve.**

A **Incorrect. While the AUTO voltage regulator has automatic circuitry to lower field current < 5876 amps, this is only activated on a generator field overvoltage condition, which has not occurred. Adjusting the manual voltage regulator to match the AUTO regulator can be performed, but will not mitigate operation outside of the capability curve.**

B **Incorrect. Placing the manual voltage regulator in MANUAL is not authorized by the procedure. There is no basis for assuming misoperation of the voltage regulator in AUTO as the stem clearly indicates the excessive reactive loading is due to grid conditions.**

C **Correct. A power reduction is authorized by ON-198-001. Performing the power reduction per the CRC instructions is the preferred method. GO-100-012 will have to be performed due to the unplanned power reduction.**

D **Incorrect. While a power reduction is authorized by ON-198-001, 5 percent is more than required to obtain a positive margin on the capability curve. Note 2 to Step 3.5.3 of ON-198-001 allows up to 2 minutes for the AUTO voltage regulator to attempt to restore margin, so immediate action is not required. The 5 percent requirement is taken from ON-193-001 for a EHC control valve oscillation.**

10CFR55	**43.5**	**This is a SRO-level question as evaluation of current generator conditions and selection of the appropriate procedure based on detailed knowledge of the mitigating strategy.**

Technical References	**ON-198-001, Section 3.5, 5.0**
Learning Objectives	**15304**
Question Source	**New**
Previous NRC Exam	**No**
Comments	Click here to enter text.

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Exam	SRO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		295005 AA2.02 Main Turban Generator Trip				Importance		2.7	
Statement		Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : Turbine vibration							

QUESTION 78

Unit 1 is shutting down for a forced outage. Reactor Power is 20 percent.

Annunciator TURB GEN BRG HI VIBRATION (AR-105-E05) alarms due to bearing #5 rotor and casing high vibration.

Operators trip the Main Turbine. The generator output breaker opens, but turbine speed does not lower.

Turbine bearing #5 vibration continues to rise. Vibration is currently 8 mils, up 1 mil every 2 minutes.

Which one of the following identifies the appropriate actions to direct to lower turbine vibration?

A. Close the MSIVs and MSL drains immediately
Verify turbine speed begins to lower

B. Place the Mode switch to SHUTDOWN immediately
Close the MSIVs and MSL drains
Verify turbine speed begins to lower

C. Place the Mode switch to SHUTDOWN immediately
Close the MSIVs and MSL drains
Open the Main Condenser vacuum breakers

D. Place the Mode switch to SHUTDOWN before #5 bearing vibration rises to 10 mils
Close the MSIVs and MSL drains
Open the Main Condenser vacuum breakers when #5 bearing vibration is > 10 mils

Proposed Answer: **B**

Applicant References: **None**

Explanation: **The Main Turbine has been tripped due to a high vibration condition. On the turbine trip leak by on the main turbine stop and control valves has resulted in the turbine remaining at speed. Turbine vibration remains high and is rising slowly.**

A Incorrect. Action to isolate steam flow to the main turbine is required by ON-193-002 Step 3.2. Although reactor power is below the bypass for reactor scram on turbine trip, directing an action, closing MSIVs, that will result in a reactor scram without first initiating a reactor scram is not allowed.

B Correct. Per ON-193-002 Step 3.2 the steam supply to the main turbine should be isolated if turbine speed does not lower after a turbine trip. Further action to break vacuum is not warranted at this time due to the slow rise in vibration.

C Incorrect. Breaking vacuum is not required until vibration is extremely high. Vibration is currently below the trip limit, so the "extremely high" threshold has not been met. This is the procedural method for breaking vacuum per ON-193-002.

D Incorrect. While the actions specified are correct and in the correct sequence, 10 mils is below the turbine trip setpoint for vibration so the CAUTION before Step 3.4 of ON 193 002 applies and action to break vacuum should be deferred until turbine speed lowers to 1200 rpm.

10CFR55	43.5
Technical References	ON-193-002 Steps 3.2, 3.4 AR-105-E05
Learning Objectives	11041
Question Source	New
Previous NRC Exam	No
Comments	

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Exam	SRO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	4
K/A		295030 G2.4.41 Low Suppression Pool Water Level				Importance		4.6	
Statement		Knowledge of the emergency action level thresholds and classifications.							

QUESTION 79

Use your provided references and the information on the next page to answer this question.

Unit 1 experienced an electrical ATWS. Initial ATWS power was 100 percent.

Subsequently, MSIVs failed closed.

Reactor level is being maintained at -130", steady, by HPCI and RCIC at full flow.

Reactor pressure is being maintained 800-1050 psig using SRVs.

All attempts at control rod movement and boron injection fail.

Subsequently, a leak occurs in the Division 1 RHR Pump room.

Operators determine that the leak is on the suction of RHR Pump 1A and cannot be isolated.

The following conditions now exist

Suppression Pool level	22 ft, down fast
Suppression Pool temperature	170 °F, up slow

Which one of the following identifies the action that will be required in response to this event, and the final Emergency Plan classification?

A. Rapid Depressurization when HCTL is violated
Site Area Emergency

B. Rapid Depressurization when reactor level falls below TAF
Site Area Emergency

C. Rapid Depressurization when HCTL is violated
General Emergency

D. Rapid Depressurization when reactor level falls below TAF
General Emergency

Exam	SRO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		**295037 G2.4.35 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown**				Importance		**4.0**	
Statement		**Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.**							

QUESTION 80

Unit 1 has experienced a failure of RPS to trip.

When ARI was initiated, a large number of control rods on the right side of the full core display continued to show not fully inserted

All actions in the power leg of EO-100-113 were completed to the point of attempting control rod insertion.

ES-158-002, ARI and RPS Trip Bypass, was directed to be performed. The in-field portion of the ES was completed.

Annunciators RPS CHAN A1/A2(B1/B2) SCRAM DSCH VOL HI WTR LEVEL TRIP (AR-103(104)-F02), have subsequently cleared.

Which one of the following should be directed next in an attempt to insert the withdrawn rods?

A. Reset the scram, then insert a manual scram using the RPS manual scram pushbuttons

B. Individually scram control rods in accordance with Attachment A of EO-100-113 Sheet 2

C. Vent the scram air header in accordance with the posted instructions

D. Insert control rods in accordance with ES-155-001, Venting CRD to Insert Control Rods

Proposed Answer **C**

Applicant References **None**

Explanation **The conditions presented in the stem are consistent with an electrical ATWS, as indicated by the failure of the full core display to enter full-in/full-out mode, where ARI initiation or maximizing CRD flow were successful in inserting most of the control rods. ES-158-002 was directed for installation to defeat ARI to re-pressurize the scram air header for subsequent scram attempts. The RPS trip bypass portion of the ES were installed, but for no effect.**

A **Incorrect. The actions described are the next steps to perform to complete ES-158-002 to attempt a re-scram. However, as RPS has failed to trip this action will not have any effect and will not insert control rods.**

B **Incorrect. Individually attempting to scram control rods may have some success, but re-venting the scram air header to attempt to re-scram all withdrawn control rods is the preferred response.**

C **Correct. With RPS untripped and ARI defeated to re-pressurize the scram air header (indicated by the SDV now being drained), this action will vent the scram air header for an attempt to re-scram the control rods.**

D Incorrect. This action would be effective in attempting to insert the control rods, but is not allowed to be used until all other methods have been attempted.

10CFR55	43.5	This question requires assessment of plant conditions (diagnosis of type of ATWS) and then selecting the appropriate procedure to continue with to make an effective attempt at control rod insertion.
Technical References	EO-100-113	
Learning Objectives	14594	
Question Source	New	
Previous NRC Exam	No	
Comments		

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Exam	SRO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		295021 G2.4.4 Loss of Shutdown Cooling				Importance		4.7	
Statement		Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.							

QUESTION 81

Unit 1 is cooling down following a scram from rated power. Reactor coolant temperature is 220 °F.

RHR Loop A is in Shutdown Cooling using RHR Pump 1A.

RWCU BLDN FLOW REG VLV, HV-144-F033, fails full open.

The ensuing level transient is terminated when RWCU isolates on low reactor level.

Which one of the following identifies the preferred course of action to re-establish decay heat removal and continue the cooldown?

A. Re-enter EO-100-102 and raise reactor level > 90" with CRD and Condensate
Perform ON-149-001 Attachment B, Quick Recovery of previously Inservice SDC Loop, to restore Division 1 RHR to Shutdown Cooling

B. Re-enter EO-100-102 and raise reactor level > 13" by realigning Division 1 RHR to LPCI
Restart a Reactor Recirc Pump per OP-164-001 Attachment D, Post Scram Recovery of A(B) Recirculation System Pump

C. Perform ON-149-001 Attachment F, Alternate Decay Heat Removal RHR Loop B Injection with Suction from the Suppression Pool

D. Raise reactor level per OP-149-002 Section 2.7, SDC Level Control Operation
If RHR Pump 1A trips, restart RHR Loop A in SDC per OP-149-002 Section 2.1, Starting RHR A(B) in SDC in Mode 3

Proposed Answer	A
Applicant References	None
Explanation	A loss of vessel level occurs due to malfunction of the RWCU blowndown valve. Reactor level falls to -38" before the level transient is terminated. As soon as RWCU isolates, CRD begins to recover level as the minimum allowed CRD injection rate per GO-100-005 Step 5.38.2.b Note. RHR SDC isolated at +13", so decay heat removal has been lost.

A Correct. Entry into EO-100-102 is required on low reactor level. Step RC/L-4 specifies an allowed level band of +90" to +100" if SDC is in operation. CRD and Condensate both remain available for vessel makeup to raise level per GO-100-005. The preferred approach to restore decay heat removal is given by ON-149-001 Step 3.3.1, which directs restoring the previously in-service RHR loop to SDC if conditions permit.

B Incorrect. EO-100-102 will allow the use of LPCI to raise level above +13", but continuous operation of an RHR pump will not be possible under EO-100-102, thus providing only intermittent decay heat removal insufficient to meet TS 3.4.. Restarting a Reactor Recirc pump would be necessary to maintain coolant circulation with limited, occasional LPCI flow.

C Incorrect. Entry into EO-100-102 is required. Performance of this ON section will eventually restore reactor level and decay heat removal, but is not preferred by EO-100-102 or ON-149-001 as RHR can be readily returned to SDC.

D Incorrect. Entry into EO-100-102 and ON-149-001 is required. Operation of RHR to restore reactor level using the referenced section of the procedure is not possible, as a RHR SDC isolation has occurred.

10CFR55	43.5	This is an SRO-level question as an assessment of plant conditions is required to idenfity the lowest reactor level reached, and selection of the appropriate procedure to restore decay heat removal is required.
Technical References	EO-100-102 ON-149-001 GO-100-005 Steps 5.35-5.38	
Learning Objectives	15304	
Question Source	New	
Previous NRC Exam	No	
Comments		

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Exam	SRO	Tier	1	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A	295019 2.1.19 Partial or Complete Loss of Instrument Air					Importance	3.8		
Statement	Ability to use plant computers to evaluate system or component status.								

QUESTION 82

Refer to the figure on the following page when answering this question.

Unit 1 was operating at rated power when the CIG 90 psig header to the Drywell isolated.

Efforts to restore CIG failed and the reactor was manually scrammed.

RPS failed to de-energize on the scram.

ARI and SLC failed to function.

Operators subsequently transition reactor level and pressure control to HPCI and SRVs.

Operators are now standing by to vent the scram air header.

Which one of the following actions will satisfy the requirements of EO-100-103, given the conditions in Containment as indicated on the plant computer, if all control rods insert when the scram air header is vented?

A. Maximize RHR Suppression Cooling per OP-149-004 to maintain operation below the HCTL limit for this reactor pressure

B. Enter EO-100-112 and perform a Rapid Depressurization due to violation of the HCTL limit

C. Lower reactor pressure regardless of cooldown rate to restore operation below the HCTL limit

D. Re-enter EO-100-103 and maximize RHR Suppression Cooling per OP-149-004 to restore operation below the HCTL limit


HEAT CAPACITY TEMP LIMIT
HEATCAP
SIM A A:
ACT B: STBY
POWER
LEVEL
PRESS
CONTN
RAD
MODE: 3
SPDS
ALARM
6/21/14 15:35:33
SPDS MENU
UNIT 1
REV: 08/19/95
MESSAGE
ALARM SUMMARY
MESSAGE SUMMARY
CONTAINMENT STATUS
DRYWELL PRESS & TEMP
CONTAINMENT H2/O2 LIMITS
CONTAINMENT PRESS LIMIT
RPV SAT TEMP LIMIT
PRESSURE SUPPRESSION LIMIT
CONTAINMENT ISOLATION
SUPP POOL TEMP (F)
191.2
260
250
240
230
220
210
200
190
180
170
160
150
12
14
16
18
20
22
24
26
28
30
0 - 95
96 - 200
201 - 400
401 - 600
601 - 800
801 - 1000
1001 - 1106
RPV PRESS (PSIG)
837
SUPP POOL WTR LVL (FT)
23.9
POWER
REACTOR LEVEL
REACTOR PRESSURE
CONT/EOP
RADIOLOGICAL RELEASE
PREVIOUS FUNCTION

Proposed Answer	**B**
Applicant References	**None**
Explanation	**An full-power isolated ATWS has occurred due to the failure of RPS and subsequent closure of MSIVs due to the loss of CIG. A R*Time display of the HCTL curve with current plant conditions is provided for use in determining the required action within EO-100-103. The conditions show operation in violation of the HCTL limit. When control rods are inserted progress in EO-100-103 can continue past SP/T-5. SP/T-8 requires a Rapid Depressurization per EO-100-112 when operation cannot be maintained within the HCTL limit.**

A **Incorrect. This is an appropriate action to initiate in response to an isolated ATWS, and is not specifically described as having been performed in the stem. However, this will not satisfy the EO-103 requirements for high SP temperature; a Rapid Depressurization will be required when all control rods are inserted.**

B **Correct. EO-103 Step SP/T-8 requires a Rapid Depressurization be initiated when HCTL cannot be maintained within limits, when all control rods are inserted.**

C **Incorrect. While this applies application of the bowtie per EO-100-102 Step RC/P-3, allowed once all control rods are inserted, once HCTL is violated a Rapid Depressurization is required by EO-103 Step SP/T-8.**

D **Incorrect. Re-entry into EO-100-102 will be required when EO-100-113 is exited when all control rods insert. Re-entry into EO-100-103 is not required. Execution of the SP temperature leg of EO-103 is stopped at step SP/T-5 with the ATWS in progress; execution of the procedure continues with Step SP/T-8 as soon as all control rods are inserted. SP/T-8 requires RD when HCTL cannot be MAINTAINED safe, no provision for violation and restoration of the limit is made.**

10CFR55	**43.5**	**This is an SRO-level question as it requires knowledge of diagnostic steps and decision points in EOP-103 that result in transition to the Rapid Depressurization EOP contingency procedure.**
Technical References	**EO-100-103** **M-126 Sht 1**	
Learning Objectives	**14622**	
Question Source	**New**	
Previous NRC Exam	**No**	
Comments	**None**	

Operations Reviewer MJ / 06/26/14
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Exam	SRO	Tier	1	Group	2	Cognitive Level	Low	Level of Difficulty	4
K/A		295020 AA2.03 Inadvertent Containment Isolation				Importance		3.7	
Statement		Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Reactor power.							

QUESTION 83

Unit 1 is operating at rated power.

The RWCU return flow instrument fails downscale.

RWCU automatically isolates.

RWCU flow on PPC OD3 display turns WHITE.

Which one of the following actions is required?

A. Enter ON-100-006, Loss of Heat Balance Calculation
Reduce core flow by 0.5 Mlbm/hr after 15 minutes

B. Enter ON-100-004, Reactor Power Greater than Authorized Limit
Immediately reduce core flow as necessary to obtain < 3952 MWth as indicated on PPC 15-minute average Core Thermal Power

C. Enter ON-156-001, Unanticipated Reactivity Change
Raise core flow as necessary to maintain PPC APRM average as close to 100 percent as possible

D. Enter ON-100-006, Loss of Heat Balance Calculation
Immediately enter a substitute value of RWCU flow of 300 gpm <u>AND</u> verify PPC 15-minute average Core Thermal Power turns YELLOW

Proposed Answer: **A**

Applicant References: **None**

Explanation: **An actual RWCU isolation has occurred due to high differential flow. In this event this has resulted in an invalid RWCU flow indication, which will result in an invalid heat balance calculation. The appropriate procedure to enter is ON-100-006 for loss of the heat balance. The necessary action within the ON is to reduce power a small amount below rated to ensure the licensed power level is not promptly violated.**

A Correct. Entry into ON-100-006 is required due to the loss of the heat balance. The appropriate response per the ON is to reduce power by reducing core flow by 0.5 Mlbm/hr.

B Incorrect. This is the correct action if core thermal power remained valid and the loss of RWCU flow would result in a rise in indicated heat balance power.

C Incorrect. Entry into ON-156-001 is not specifically required for a loss of RWCU flow as none of the symptoms include loss of RWCU flow or the heat balance. The procedure does not address loss of the heat balance or loss of RWCU flow. Raising core flow when the heat balance has been lost violates the guidance of ON-100-006.

D **Incorrect. While entry into ON-100-006 is required, action to substitute a RWCU flow is not immediately required. Consultation with Reactor Engineering is required. Use of 300 gpm as a substitute value is significantly over-conservative as actual RWCU flow is 0 gpm.**

10CFR55 **43.5** **This questions is at the SRO level as assessment of plant conditions to identify why the RWCU flow input to the heat balance was lost (isolation, as opposed to failure of the return flow which does not input to the HB) and selection of the appropriate procedure to mitigate the loss of the heat balance.**

Technical References	**ON-100-006**
Learning Objectives	**15304**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

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Init / date

Exam	SRO	Tier	1	Group	2	Cognitive Level	High	Level of Difficulty	3
K/A		295002 G2.4.21 Loss of Main Condenser Vacuum				Importance		4.6	
Statement		Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.							

QUESTION 84

Unit 1 was manually scrammed due to Main Condenser air in-leakage.

All Feedwater pumps tripped on low vacuum after aligning to startup level control.

When HPCI was initiated for reactor level control, an unisolable steam leak in the HPCI room occurred.

The leak has resulted in temperatures in both the HPCI and RCIC pump rooms rising.

All actions in EO-100-104 to mitigate the effects of the steam leak have been attempted.

HPCI and RCIC room temperatures continue to rise and are approaching Maximum Safe values.

Reactor pressure is being maintained at 935 psig by Main Turbine Bypass valves.

Which one of the following describes the most rapid method of lowering reactor pressure allowed by Emergency Operating procedures for this condition?

A. Enter EO-100-112 for Rapid Depressurization and open 6 ADS/SRVs

B. Open 6 ADS/SRVs to depressurize regardless of cooldown rate

C. Fully open all Main Turbine bypass valves to depressurize regardless of cooldown rate

D. Open SRVs to reduce reactor pressure to 450-600 psig

Proposed Answer	C
Applicant References	None
Explanation	A primary system is discharging to the Secondary Containment and cannot be isolated. Two areas of Secondary Containment are approaching the Maximum Safe temperature. Per EO-100-104 step SC/T-8, a Rapid Depressurization will be required when both room temperatures exceed Max Safe. RD is imminent as the steam leak is unisolable and temperatures continue to rise. EO-100-102 step RC/P-3 allows cooldown in excess of the TS limit when RD is anticipated. In this condition, even though condenser vacuum is degrading, EO-100-102 step RC/P-3 prefers directing as much energy as possible to a heat sink other than the Suppression Pool. The only requirements for use of the bypass valves to anticipate Rapid Depressurization is that the bypass valves be operable with an unisolated MSL and the Main Condenser still in service.

A Incorrect. Rapid Depressurization per EO-100-104 SC/T-8 is not required until 2 Secondary Containment area temperatures exceed Max Safe. That has not yet happened.

B Incorrect. Use of SRVs to anticipate Rapid Depressurization per EO-100-102 is not allowed, only use of the bypass valves is authorized.

C Correct. This is the preferred method of utilizing a heat sink other than the Suppression Pool in anticipation of a Rapid Depressurization.

D Incorrect. While this method of pressure control is allowed by EO-100-102 Step RC/P-6 to allow injection from Condensate, discharging as much energy to a heat sink other than the Suppression Pool is preferred.

10CFR55	43.5
Technical References	EO-100-102 Step RC/P-3 EO-100-104 Step SC/T-8
Learning Objectives	14624
Question Source	New
Previous NRC Exam	No
Comments	

Operations Reviewer mj / 06/03/2014
Init / date

Facility Representative ____/______
Init / date

Exam	SRO	Tier	1	Group	2	Cognitive Level	High	Level of Difficulty	4
K/A	295022 AA2.03 Loss of CRD Pumps					Importance	3.2		
Statement	Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: CRD mechanism temperatures								

QUESTION 85

Use your provided references to answer this question.

Unit 1 is operating at rated power when the in-service CRD Pump trips on low suction pressure due to a pump suction filter high ΔP condition.

Operators are dispatched to bypass the CRD pump suction filter per ON-155-007, Loss of CRD System Flow.

The following alarms are subsequently received

CRD PANEL 1C007 HI TEMP (AR-103-H05)
CRD ACCUMULATOR TROUBLE (AR-103-H06)

Abnormal conditions are noted for the four Control Rods as shown below, ONLY.

15	270 1050 48	355 950 00
11	380 950 48	295 925 48
	22	26

CRDM temp (°F) HCU accum press (psig) Control rod position

Which one of the following identifies the action(s) and latest completion time(s) that will satisfy ALL Technical Specifications requirements for this condition?

A. Restore Control Rod 26-11 HCU accumulator pressure within 8 hours
Restore Control Rod 22-11 OR 26-15 CRDM temperature within 12 hours

B. Restore Control Rod 26-11 HCU accumulator pressure within 1 hour
Be in MODE 3 in 12 hours regardless of CRDM temperatures

C. Restore Control Rod 26-11 HCU accumulator pressure within 20 minutes
Be in MODE 3 in 12 hours regardless of CRDM temperatures

D. Declare Control Rod 22-11 INOPERABLE within 12 hours, ONLY

Proposed Answer: A

Applicant References: **TS 3.1.4 (redacted)**
TS 3.1.5 (redacted)

Explanation

Both CRD Pumps are unavailable due to a low suction pressure condition induced by high dP across the common pump suction filter. The loss of CRD cooling water flow will result in elevated temperatures in the CRD mechanisms eventually resulting in CRDM temperatures over 350 °F, the OI-055-003 limit requiring control rods to be declared SLOW. Similarly the loss of charging water header pressure will result in individual HCU accumulator pressure falling below the TS SR3.1.5.1 operability limit of 940 psig.

A **Correct. TS 3.1.5 Condition A applies for a single inoperable HCU accumulator. If accumulator pressure is restored within the 8-hour Completion Time for either Required Action A.1 or A.2 that LCO is met and no additional action is required by TS 3.1.5, per LCO 3.0.2. Declaring either of Control Rods 22-11 or 26-15 INOPERABLE satisfies the total number and separation criteria of TS 3.1.4, in that only 1 OPERABLE control rod is SLOW, and no further action would be required per LCO 3.0.2 as the TS 3.1.5 LCO is met.**

B **Incorrect. 1 hour is the Completion Time for 2 or more inoperable HCU accumulators from TS 3.1.5 Condition B. CRDM temperatures are expected to be restored with a CRD pump and OI-055-003 only requires declaring the control rod slow while CRDM temperature is > 350 °F.**

C **Incorrect. 20 minutes is the Completion Time for restoring CRD charging water header pressure with 2 control rod accumulators inoperable and is not associated with restoring HCU accumulator pressure. CRDM temperatures are expected to be restored with a CRD pump and OI-055-003 only requires declaring the control rod slow while CRDM temperature is > 350 °F.**

D **Incorrect. While declaring control rod 22-11 INOPERABLE will satisfy LCO 3.1.4, in that no SLOW rod is adjacent to another OPERABLE SLOW rod, HCU accumulator pressure for control rod 26-11 renders that accumulator inoperable and action is required to satisfy TS 3.1.5.**

10CFR55 **43.2** **This question is SRO-level in that it requires determination of Required Action with Completion Times greater than 1 hour.**

Technical References **OI-055-003, Section 4.6**
TS 3.1.4
TS 3.1.5

Learning Objectives **13112**

Question Source **New**

Previous NRC Exam **No**

Comments

Operations Reviewer mj / 05/06/14
Init / date

Facility Representative ____/______
Init / date

Exam	SRO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A		263000 A2.02 D.C. Electrical Distribution				Importance		2.9	
Statement		Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of ventilation during charging							

QUESTION 86

Both Units are operating at rated power.

Battery Charger 1D663 is in EQUALIZE per Maintenance request. All other 125/250V DC battery chargers are in FLOAT.

Battery Room Exhaust Fan OV116A trips. Standby fan 0V116B fails to start.

Which one of the following describes the actions to be directed for the loss of battery room ventilation per ON-030-002?

A. Place Battery Charger 1D663 in FLOAT within 3 hours

B. Open Unit 1 and 2 Battery Room doors
Enter TRM LCO 3.7.3.7 for inoperable fire doors

C. Place CREOASS in service in Pressurization/Filtration Mode per OP-030-002
Open Unit 1 and 2 Battery Room doors
Enter TRM LCO 3.7.3.7 for inoperable fire doors

D. Place CREOASS in service in Pressurization/Filtration Mode per OP-030-002
Open Unit 1 and 2 Battery Room doors
Enter TRM LCO 3.7.3.7 for inoperable fire doors
Place Battery Charger 1D663 in FLOAT within 3 hours

Proposed Answer: **D**

Applicant References: **None**

Explanation: **Both divisions of Battery Room exhaust ventilation have been lost. One division of battery Room exhaust ventilation is required by TRO 3.7.9 for operability of the equipment in the 125/250V DC battery rooms for cooling and combustible gas control.**

Restoration of flow through the battery rooms is required to prevent buildup of combustible gases in the battery rooms. This is accomplished by starting CREOASS in the PRESSURIZATION/FILTRATION mode to bring in fresh air from the CS intake and circulate it through the Control Structure. Opening the battery room doors allows air circulation through the battery room sufficient for cooling and dissipating hydrogen.

A Placing the 1D663 charger in FLOAT limits hydrogen production from the charging 1D660 battery, but hydrogen is still being produced from all batteries due to operation of the associated chargers in FLOAT mode and building up in the isolated battery room spaces.

B Incorrect. Opening the doors to the battery rooms will allow airflow from the normal CS HVAC to enter the battery rooms, but hydrogen is still being generated from the batteries and will rise in concentration in the highest elevations of the Control Structure. A purge of the CS airspace is required to limit hydrogen buildup. This is the correct TRM LCO for an inoperable fire door.

C Incorrect. Operation of CREOASS in the PRESSURIZATION/FILTRATION mode will result in a constant feed and bleed on the CS airspace, limiting hydrogen buildup. However, action to place all battery chargers in FLOAT is still required to limit hydrogen generation.

D Correct. Operation of CREOASS in the PRESSURIZATION/FILTRATION mode will result in a constant feed and bleed on the CS airspace, limiting hydrogen buildup. Placing all battery chargers in FLOAT limits hydrogen generation to the minimum possible.

10CFR55 43.5 This is an SRO-level question as plant conditions must be evaluated to determine the effect of the isolation on battery room ventilation, the correct procedure selected to respond to the loss of ventilation, and application of license requirements for Appendix R compliance.

Technical References
ON-030-002 Section 3.4, 5.0
OP-030-002 Section 2.10
TRO 3.7.9
TRM 3.7.3.7

Learning Objectives 10455

Question Source Bank LXR ILO TMOP401/13058/003

Previous NRC Exam No

Comments

Operations Reviewer mj / 06/23/14
Init / date

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Exam	SRO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A		**209001 A2.01 Low Pressure Core Spray**				Importance		3.4	
Statement		**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: Pump trips**							

QUESTION 87

Unit 1 experienced a large-break LOCA at rated power.

Only Division 2 ECCS systems are available.

Reactor level was recovered with injection from Core Spray and RHR in the LPCI mode.

RHR Loop B has been aligned to Drywell and Suppression Chamber spray.

Core Spray Loop B maintained reactor level -140", steady, on Compensated Fuel Zone.

Core Spray Pump 1D then tripped.

Reactor level is now -200", steady, on Compensated Fuel Zone.

Which one of the following describes the next required action per Emergency Operating Procedures to assure adequate core cooling?

A. Initiate a Rapid Depressurization

B. Initiate a Rapid Depressurization AND direct Core Spray Loop B flow throttled to < 3950 gpm

C. Direct RHR Loop B re-aligned for LPCI injection to restore reactor level with flow through the RHR heat exchanger

D. Contact the TSC to enter EP-DS-002 for RPV and Primary Containment flooding

Proposed Answer **C**

Applicant References **None**

Explanation **A large-break LOCA with degraded ECCS response will prevent completely recovering level in the RPV. The initial conditions in the stem are consistent with the long-term response to a DBA LOCA. The loss of Core Spray flow will result in level inside the shroud lowering and resulting in a loss of adequate core cooling by submergence. With 1 CS pump tripped adequate core cooling by spray does not exists. RC/L-21 requires maximizing RPV injection under these conditions.**

A Incorrect. Conditions are already met for an automatic ADS initiation, with level < -129" for sufficient time elapsed after the LOCA to allow RHR to be realigned for containment cooling.

B Incorrect. Rapid Depressurization will have already occurred. The action to throttle Core Spray flow is required by OP-151-001.

C Correct. Realigning RHR to LPCI is the next required action in response to the loss of adequate core cooling as the override at RC/L-19 must now be answered NO in response to the loss of design CS flow. Use of RHR for LPCI per Table 3 prompts directing flow through the RHR HX as soon as possible. In this condition, 10,000 gpm of RHR flow should be adequate to restore and maintain RPV level.

D Incorrect. The decision to enter RPV and PC flooding is not required until a determination is made that level cannot be restored and maintained > TAF. With Core Spray able to maintain level > TAF before a pump tripped, RHR will also be able to maintain level > TAF.

10CFR55	43.5	This questions is SRO-level because knowledge of the diagnostic steps of the Alternate Level Control contingency EOP is required.
Technical References	EO-102	
Learning Objectives	14622	
Question Source	New	
Previous NRC Exam	No	
Comments		

Operations Reviewer mj / 05/06/14
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Exam	SRO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	4
K/A	262001 G2.1.32 A.C. Electrical Distribution					Importance	4.0		
Statement	Ability to explain and apply system limits and precautions.								

QUESTION 88

Unit 1 is in Mode 4 for a refueling outage. A Division 2 outage window is in progress.

Unit 2 is operating at rated power.

0ATS526 is overheating due to a bad contactor. It has been removed from service to allow repairs. All loads supplied from 0ATS526 are de-energized.

TS LCOs

TS 3.5.1 Emergency Core Cooling Systems – Operating
TS 3.8.4 DC Sources – Operating
TS 3.8.7 Electrical Distribution – Operating

Which one of the following describes the Technical Specification LCO entry requirements for the DC systems affected on Unit 2 for this condition?

A. Enter TS 3.8.4 for 1 required DC battery charger inoperable
No safety function determination per LCO 3.0.6 is required

B. Enter TS 3.8.4 for 2 required DC battery chargers inoperable
Perform a safety function determination per LCO 3.0.6
No loss of safety function exists

C. Enter TS 3.8.7 for 1 required DC distribution system inoperable
No safety function determination per LCO 3.0.6 is required

D. Enter TS 3.8.7 for 2 required DC distribution systems inoperable
Perform a safety function determination per LCO 3.0.6
Enter LCO 3.0.3 for a loss of safety function in TS 3.5.1

Proposed Answer **B**

Applicant References **None**

Explanation

The 0ATS526 is the normal supply to Division 2 ESS 480V LC 0B526. To perform maintenance on the ATS the normal and alternate power supplies must first be de-energized. OP-105-001 Section 2.10 is the procedure governing this activity when performed for scheduled maintenance.

0B526 is the power supply to the Division 2 125V DC battery chargers on Unit 1 and 2. The chargers will be de-energized when 0B526 is de-energized. The associated DC buses 1D620 and 2D620 remain operable with the associated batteries operable.

NDAP-QA-0312 Att B defines the systems to which LCO 3.0.6 is applicable. DC sources are identified as a support system to the DC distribution systems required operable by TS 3.8.7.

A Incorrect. A safety function determination is required per NDAP-QA-0312. This distractor is plausible in that a specific exception to application of LCO 3.0.6 is made for TS 3.8.1 AC Sources, but not for TS 3.8.4 DC Sources. Specification of 1 DC source is plausible as the Unit 1 charger is required for Unit 2 operation, although not for Unit 1.

B Correct. Both the Unit 1 and 2 battery chargers are required to be operable for Unit 2 by TS 3.8.4. A safety function determination is required by NDAP-QA-0312.

C Incorrect. LCO 3.0.6 is applicable to the inoperable battery charger and entry into LCO 3.8.7 is not required. Specification of 1 DC distribution system is plausible as the Unit 1 DC distribution system 1D620 is required for Unit 2 operation, although not for Unit 1.

D Incorrect. LCO 3.0.6 is applicable to the inoperable battery charger and entry into LCO 3.8.7 is not required. A determination of a loss of safety function is plausible in that if LCO 3.0.6 is not applied with TS 3.8.7 a loss of safety function in TS 3.5.1 does exist due to inoperability of Division 2 of Core Spray and RHR due to inoperable logic and breaker control power supplies, among other systems.

10CFR55	43.2	This is an SRO-level question as determining the correct answer requires application of generic LCO requirements regarding safety function determination.
Technical References	OP-105-001 Section 2.10 TS 3.8.4 NDAP-QA-0312	
Learning Objectives	10976	
Question Source	New	
Previous NRC Exam	No	
Comments	This question satisfies the K&A as the examinee is required to apply P&L 2.10.2e of OP-105-001 and identify the specific LCOs that are applicable while the ATS is removed from service.	

Operations Reviewer mj / 05/06/14
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Exam	SRO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	2
K/A		223002 2.2.40 Primary Containment Isolation System / Nuclear Steam Supply Shut-Off				Importance		4.7	
Statement		Ability to apply Technical Specifications for a system.							

QUESTION 89

Use your provided references when answering this question.

Unit 1 is operating at rated power.

PDIS-G33-1N044B, RWCU B System High Flow, fails downscale.

Which one of the following identifies the action required by Technical Specifications for this condition?

A. Restore isolation capability or place the channel in trip within 24 hours
<u>IF</u> isolation capability is not restored, isolate RWCU within the following 1 hour

B. Isolate RWCU within 4 hours

C. Isolate RWCU within 1 hour

D. Restore isolation capability or place the channel in trip within 1 hour
<u>IF</u> isolation capability is not restored, isolate RWCU within the following 1 hour

Proposed Answer: **A**

Applicant References: **TS 3.3.6.1 (partial)**
TS 3.6.1.3

Explanation: **PDIS-G33-1N044B is the RWCU system flow transmitter that provides the signal for the TS 3.3.6.1 Function 5.g isolation. Failure of the transmitter downscale renders the trip capability of the B trip channel lost and RWCU inlet O/B isolation valve HV-144-F004 will not close on a valid high-flow condition. Isolation capability for Function 5.g is not lost, as the A trip channel will automatically close the HV-144-F001 valve accomplishing the isolation function.**

- **A Correct. Loss of the trip capability in 1 trip channel for 24 hours is allowed by TS 3.3.6.1 Condition A.**
- **B Incorrect. This is the Required Action and Completion Time for an inoperable PCIV per TS 3.6.1.3 Condition A.**
- **C Incorrect. This is the Required Action and Completion Time for both PCIVs inoperable in a penetration per TS 3.6.1.3 Condition B.**
- **D Incorrect. This is the Required Action and Completion Time for a loss of isolation capability for Function 5.g. The isolation capability of the A trip channel is maintained.**

10CFR55: **43.2** **This is an SRO-level question as it requires application of TS Required Actions > with Completion Times > 1 hour.**

Technical References: **TS 3.3.6.1**
M1-B21-131 Sht 9

Learning Objectives: **13180**

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Question Source	Bank	LXR LOR TMOP061/16180/5
Previous NRC Exam	No	
Comments		

Operations Reviewer MJ / 03Jun14
Init / date

Facility Representative ____/______
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SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Exam	SRO	Tier	2	Group	1	Cognitive Level	Low	Level of Difficulty	4
K/A		400000 Component Cooling Water				Importance		4.6	
Statement		Ability to interpret and execute procedure steps.							

QUESTION 90

Unit 1 is operating at rated power. Unit 2 is starting up from a refueling outage, preparing to enter Mode 1.

A leak develops on the ESW supply piping to Diesel Generator C

Due to the magnitude and location of the leak, both loops of ESW to DG C are isolated.

Subsequently, it is determined that the leak only affects the ESW Loop A supply line to DG C.

Which one of the following identifies the action(s) required, if any, to satisfy Technical Specification LCO 3.0.4 requirements to allow Unit 2 to enter Mode 1?

A. Unit 2 may enter Mode 1 without any other action as only 1 Technical Specification-required system is inoperable

B. Perform a risk evaluation of 1 required ESW subsystem inoperable and implement the associated risk management actions

C. Perform a risk evaluation of 1 required ESW subsystem inoperable AND 1 required DG inoperable and implement the associated risk management actions

D. Realign ESW Loop B to DG C per OP-054-001, ESW System
OR
Substitute DG E for DG C per OP-024-004, Transfer and Test Mode Operations of Diesel Generator E

Proposed Answer D

Applicant References None

Explanation **In the condition described, DG C has been made inoperable due to a leak in ESW with both loops to the DG isolated. A mode change is pending on Unit 2. The Note to TS 3.7.2 Conditions requires entry into LCO 3.8.1 for DGs made inoperable by inoperable ESW. With the DG inoperable, the Note to LCO 3.8.1 Conditions prohibits the use of LCO 3.0.4b risk-informed Mode changes for inoperable DG. NDAP-QA-1902 Step 6.8.2a states the same requirement.**

A **Incorrect. This represents mis-application of NDAP-QA-1902 Step 6.8.4. While this step would be applicable for 1 ESW subsystem inoperable, it may not be utilized when LCO 3.0.4b is prohibited, as is the case for an inoperable DG.**

B **Incorrect. This represents mis-application of NDAP-QA-1902 Step 6.8.5 and failure to correctly apply Step 6.8.2a.**

C **Incorrect. This represents mis-application of NDAP-QA-1902 Step 6.8.5 and failure to correctly apply Step 6.8.2a.**

D Correct. DG C must be restored to OPERABILITY or substituted with an OPERABLE DG E to allow the mode change, to satisfy NDAP-QA-1902 Step 6.8.2a.

10CFR55	43.2	This is an SRO-level question as it requires the application of generic LCO requirements (LCO 3.0.4).
Technical References	ON-054-001 Step 3.4.8, 3.5 NDAP-QA-1902 Step 6.8 TS 3.8.1 TS 3.7.2	
Learning Objectives	13426	
Question Source	New	
Previous NRC Exam	No	
Comments		

Operations Reviewer NJ / 03Jun14
Init / date

Facility Representative ____/____
Init / date

Exam	SRO	Tier	2	Group	2	Cognitive Level	High	Level of Difficulty	2
K/A		**226001 A2.11 RHR/LPCI: Containment Spray System Mode**				Importance		**3.0**	
Statement		**Ability to (a) predict the impacts of the following on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Motor operated valve failures.**							

QUESTION 91

Refer to the figure on the following page when answering this question.

Unit 1 experienced a LOCA in the Drywell at rated power.

The reactor automatically scrammed.

EO-100-103 was entered and RHR was aligned as follows:

RHR Loop A	Suppression Chamber spray
RHR Loop B	Suppression Pool cooling

Subsequently, Drywell pressure continued to rise and Drywell spray was required.

When operators attempted to align RHR Loop A for Drywell spray, power to HV-151-F016A, DRYWELL SPRAY OB ISO, was lost.

Current containment conditions are as follows:

Drywell pressure	28 psig, up slow
Suppression Chamber pressure	25 psig, up slow
Suppression Pool level	25 ft, down slow

Which of the following should be directed in response to the failure of the RHR A Drywell spray valve, in accordance with EO-100-103?

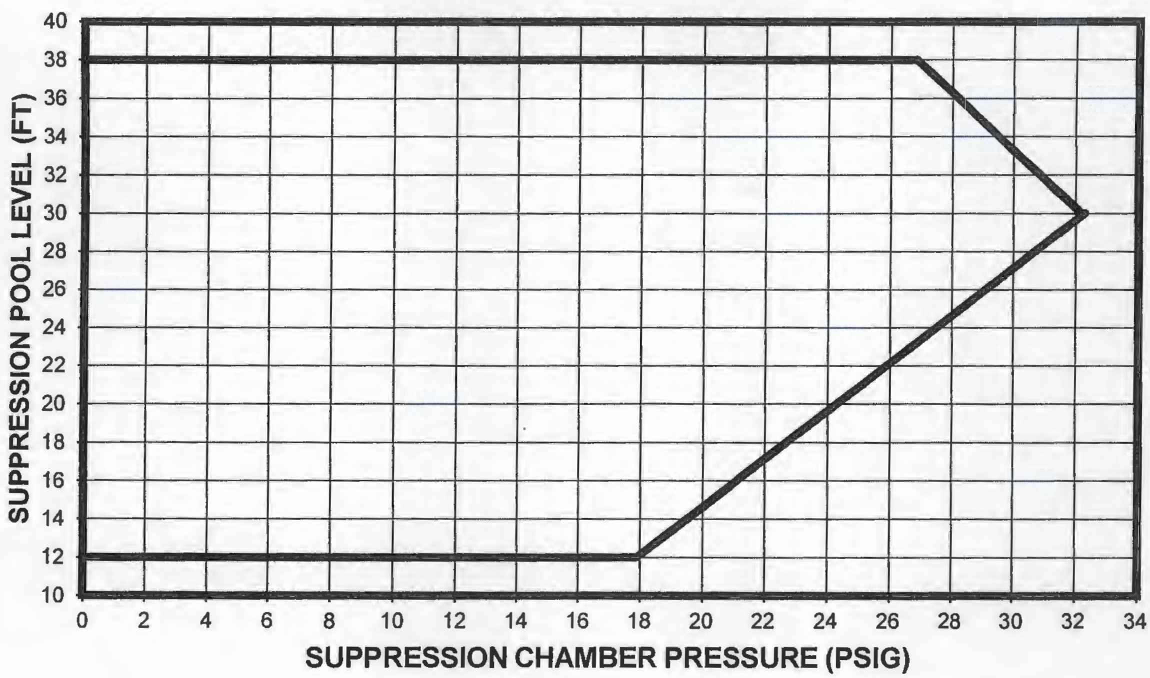
A. Immediately perform EO-100-112, Rapid Depressurization due to containment pressure exceeding the Pressure Suppression Limit

B. Direct a local operator to fully open HV-151-F016A, as sufficient Drywell overpressure exists to preclude exceeding the Drywell negative pressure limit

C. Re-align RHR Loop B from Suppression Pool cooling to Drywell spray per OP-149-004, to maximize Drywell pressure reduction

D. Place RHR Loop B in Drywell spray per OP-149-004, limiting flow through each flow path to < 10,000 gpm, to maximize decay heat removal

PRESSURE SUPPRESSION LIMIT



Proposed Answer: C

Applicant References: None

Explanation:

Primary containment is being challenged during a Drywell LOCA condition on Unit 1. Drywell pressure has risen above 13 psig and is continuing to rise to approach the PSL limit. Current conditions remain safe on the PSL curve. Attempts to spray the Drywell are appropriate before initiating RD due to approaching the PSL curve.

When RHR A was being placed in service the OB DW spray valve failed. This is the throttle valve in the DW spray flowpath used to limit initial DW spray flow to prevent damage to the primary containment due to excessive negative pressure during the initial evaporative cooling phase. Failure of this valve precludes placing RHR A in service in DW spray per procedure.

A Incorrect. This is the action required by EO-100-103 if Suppression Chamber pressure exceeds the PSL limit. Drywell pressure above the PSL limit does not require any action.

B Incorrect. Fully opening the RHR A F016A valve does not allow establishing 1000-2800 gpm flow for the first 30 seconds of DW spray operation, as SSES does not provide a DWSIPL curve.

C Correct. This action is allowed by EO-100-103 and OP-149-004. This will maximize the DW pressure reduction and if possible prevent exceeding the PSL limit.

D Incorrect. While placing RHR in both the SP cooling and DW spray modes is allowed by the note to Step 2.1 of OP-149-004, the RHR HX flow limit of 10,000 gpm still applies. No limit on total system flow of 20,000 gpm exists.

10CFR55: 43.5 This is an SRO-level question as it requires assessment of the availability of RHR A for the DW spray function due to the MOV loss and selection of the appropriate procedure to implement in response to the rising containment pressure and degraded RHR A system.

Technical References:
OP-149-004 Sect 2.1
EO-100-103 Step PC/P-7,8,9

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Learning Objectives	**10772**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer MJ / 03JUN14
Init / date

Facility Representative ____/______
Init / date

Exam	SRO	Tier	2	Group	2	Cognitive Level	High	Level of Difficulty	2
K/A		241000 2.4.34 Reactor/Turbine Pressure Regulating System				Importance		4.1	
Statement		Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.							

QUESTION 92

Unit 1 is operating at rated power.

A fire develops in the 1C651 panel. Turbine pressure set begins to lower uncontrollably as a result.

Operators place the Mode switch in SHUTDOWN and verify all control rods insert.

Control Room evacuation is ordered due to heavy smoke and flames. Immediate operator actions are NOT performed.

Which one of the following identifies the sequence of actions to be performed in response to the effects of the fire on Main Turbine EHC per ON-100-009, Control Room Evacuation?

A. Open RPS breakers CB2A and CB8B at 1Y201A and 1Y201B
Direct closure of all HV-10603A(B)(C), RFP A(B)(C) DSCH ISO
Transfer control to Remote Shutdown Panel
Transfer both HS-54101A(B), MSIV LOGIC A(B) POWER SUPPLY, to EMERGENCY

B. Transfer control to Remote Shutdown Panel
Transfer both HS-54101A(B), MSIV LOGIC A(B) POWER SUPPLY, to EMERGENCY
Open RPS breakers CB2A and CB8B at 1Y201A and 1Y201B
Direct closure of all HV-10603A(B)(C), RFP A(B)(C) DSCH ISO

C. Transfer control to Remote Shutdown Panel
Transfer both HS-54101A(B), MSIV LOGIC A(B) POWER SUPPLY, to EMERGENCY
Open RPS breakers CB2A and CB8B at 1Y201A and 1Y201B

D. Direct closure of all HV-10603A(B)(C), RFP A(B)(C) DSCH ISO
Transfer control to Remote Shutdown Panel
Transfer both HS-54101A(B), MSIV LOGIC A(B) POWER SUPPLY, to EMERGENCY

Proposed Answer A

Applicant References None

Explanation

For a control room fire ON-100-109 specifies the action to take to mitigate the possible spurious operation of Control Room equipment. In the postulated scenario reactor pressure will be lowering due to the misoperation of the Main Turbine pressure regulator due to a fire-induced malfunction of the pressure regulator setpoint. While the ON allows deviation from the order of sections of the procedure, in this scenario such deviation is inappropriate due to the pressure reduction transient in progress.

The first action required to stop the uncontrolled reduction in reactor pressure is to close the MSIVs. OP-AD-055 Step 8.6.10.b addresses pressure control when EOPs are entered and allows operator action to terminate pressure reduction to maintain pressure > 800 psig.

EO-100-102 Step RC/P-1 requires action to prevent uncontrolled condensate injection before reactor pressure < 700 psig. The method for accomplishing this per ON-100-009 is to close the RFP discharge isolation valves.

To stabilize the plant control should then be transferred to the RSDP. Subsequent action to ensure spurious re-opening of the MSIVs is required by the procedure, but should be prioritized last due to the de-energization of the RPS power supply to the MSIV solenoids.

A **Correct. This is the preferred sequence of events to respond to a fire-induced malfunction of turbine pressure control.**

B **Incorrect. While ON-100-009 allows performing sub-sections out of order, in this event this sequence transferring control to the RSDP before taking action to close the MSIVs and prevent uncontrolled injection from Condensate would contradict the guidance of EO-100-102.**

C **Incorrect. This sequence transferring control to the RSDP before taking action to close the MSIVs allows the reactor pressure reduction to continue for a longer duration. Transferring the MSIV power supply HS to EMERGENCY closes the MSIVs, additional action to de-energize RPS circuitry is not required as all control rods inserted on the scram.**

D **Incorrect. This sequence does include action to prevent uncontrolled injection from Condensate, but transferring control to the RSDP before taking action to close the MSIVs does not reflect the preferred sequence specified by ON-100-009 and allows the reactor pressure reduction to continue for a longer duration.**

10CFR55 **43.5** **This is an SRO-level question as the priority for local action is required to be selected and the effects of the local actions are required to be evaluated to select the correct response. Detailed sequencing of activities by the SRO is required to ensure control of reactor pressure and level is promptly established.**

Technical References	**ON-100-109 Section 3, 4.2-4.3** **OP-AD-055 Step 8.5.6**
Learning Objectives	**15304**
Question Source	**New**
Previous NRC Exam	**No**
Comments	

Operations Reviewer mj / 06/05/14 **Facility Representative** ____/______
Init / date **Init / date**

Exam	SRO	Tier	2	Group	2	Cognitive Level	Low	Level of Difficulty	2
K/A		234000 2.2.12 Fuel Handling Equipment				Importance		4.1	
Statement		Knowledge of surveillance procedures.							

QUESTION 93

Refer to the information on the following page when answering this question.

Unit 1 is in a refueling outage. Preparations for in-vessel fuel movement are in progress.

The status of the Refuel Platform main hoist surveillances is as follows

Procedure	Title	TS/TRM SR Satisfied	Last Performed
SO-181-001	Weekly Unit 1 Refueling Platform Grapple Operability	SR 3.9.1.1	August 14 at 1200
SO-181-004	Outage Unit 1 Refueling Platform Grapple Operability	TRS 3.9.3.1	August 15 at 1200

Initial core offload is scheduled to begin on August 22 at 1800.

Which one of the following identifies only those Refueling Platform surveillances that must be re-performed before in-vessel fuel movement can begin per the schedule, per NDAP-QA-0722, Surveillance Testing Program?

A. No surveillances are required to be re-performed

B. Perform SO-181-001, ONLY

C. Perform SO-181-004, ONLY

D. Perform SO-181-001 AND SO-181-004

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Perform CHANNEL FUNCTIONAL TEST on each of the following required refueling equipment interlock inputs: a. All rods in, b. Refuel platform position, c. Refuel platform fuel grapple, fuel loaded, d. Refuel platform frame mounted hoist, fuel loaded, e. Refuel platform monorail mounted hoist, fuel loaded.	7 days
TRS 3.9.3.1	Demonstrate the refueling platform main hoist used for handling of control rods or fuel assemblies within the reactor pressure vessel to be OPERABLE	Within 7 days prior to the start of such operations

Proposed Answer: D

Applicant References: None

Explanation: The applicant is required to identify whether Refueling Platform surveillances are current prior to initial in-vessel fuel movement activities. The applicant must apply SR applicability guidance in TS SR 3.0.2 and TRM TRS 3.0.2.

SO-181-001 is the SO used to satisfy the requirements of TS SR 3.9.1.1. The SO will have been last performed 8.25 days ago at the scheduled time for fuel movement. The TS SR 3.0.2 grace of 1.25 times the SR frequency applies (8.75 days), so the SO is not required to be performed again until 0600 on August 23 if the grace is to be applied. NDAP-QA-0722 states that the station expectation is that all routine surveillance activities will be performed without reliance on the use of grace. As SO-181-004 will have to be performed, deferring performance of SO-181-001 to use the grace would contradict the expectation set by the procedure.

SO-181-004 is the SO used to satisfy the requirements of TRM TRS 3.9.3.1. The SO will have been last performed 7.25 days ago at the scheduled time for fuel movement. TRM TRS 3.0.2 does not allow application of a grace period for TRS with a frequency of "once", which is true of TRS 3.9.3.1.

A Incorrect. Both surveillances must be re-performed prior to fuel movement.

B Incorrect. Both surveillances must be re-performed prior to fuel movement.

C Incorrect. Both surveillances must be re-performed prior to fuel movement.

D Correct. Both surveillances must be re-performed prior to fuel movement to satisfy TS/TRM and the station expectation set forth in NDAP-QA-0722 Step 7.1.6.b.

10CFR55: 43.2 This is an SRO-level question because application of generic LCO requirements (SR 3.0.2) is required.

Technical References:
TS SR 3.0.2
TRM TRS 3.0.2
TS 3.9.1
TRM 3.9.3
NDAP-QA-0722 Step 7.1.6.b

Learning Objectives: 13386

Question Source: New

Previous NRC Exam: No

Comments

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Operations Reviewer mj / 06/26/14
Init / date

Facility Representative ____/______
Init / date

Exam	SRO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	3
K/A	2.1.34 Conduct of Operations					Importance	3.5		
Statement	Knowledge of primary and secondary plant chemistry limits.								

QUESTION 94

Use your provided references when answering this question.

Unit 1 is operating at 20 percent power when Chemistry reports the following reactor coolant parameters to the Control Room.

Conductivity	11 µmho/cm
Chlorides	0.300 ppm
pH	8.8

After 6 hours, reactor power has been lowered and Mode 2 has been entered. The following reactor coolant parameters are reported:

Conductivity	0.9 µmhos/cm
Chlorides	0.150 ppm
pH	6.5

Which one of the following describes the actions to be taken?

A. Restore chlorides to within limits in the next 66 hours
Verify the cumulative time exceeding the limit is ≤ 336 hours in the past year

B. Restore chlorides to within limits in the next 48 hours
OR
Be in Mode 3 in the following 12 hours AND Mode 4 in the following 36 hours

C. Be in Mode 3 in the next 12 hours AND in Mode 4 in the next 36 hours

D. Be in Mode 3 in the next 6 hours AND in Mode 4 in the next 30 hours

Proposed Answer: D

Applicant References: TRM 3.4.1

Explanation: **Initially pH, conductivity, chloride levels are all out of specification per TRM 3.4.1. With conductivity above 10 µmho/cm Condition B is not applicable and Condition E is applicable. The Note on Condition E requires completion of the Required Actions once the condition is entered. Therefore Unit 1 must be in Mode 3 within 12 hours of the initial chemistry excursion and in Mode 4 within 36 hours of the initial excursion.**

A **Incorrect. This reflects application of Condition B for conductivity and chlorides, which is not allowed, and Condition C for pH.**

B **Incorrect. This reflects application of Condition F for Mode 2 operations. As Condition E was entered its Required Actions are more limiting.**

C **Incorrect. This reflects application of Condition E at the current time, not for 6 hours previous.**

	D	Correct. This is the correct application of Condition E Required Actions and Completion Times.
10CFR55	43.2	This is an SRO-level question as it requires application of Required Actions and Completion Times.
Technical References	TRM 3.4.1	
Learning Objectives		
Question Source	Bank	
Previous NRC Exam	Yes	LOC23
Comments		

Operations Reviewer MW / 03Jun14
Init / date

Facility Representative ____/______
Init / date

Exam	SRO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	3
K/A	2.2.23 Equipment Control					Importance	4.6		
Statement	Ability to track Technical Specification limiting conditions for operations.								

QUESTION 95

Which one of the following identifies a condition where application of the Maximum Out Of Service Time is required by NDAP-QA-0312, Control of LCOs, TROs, and Safety Function Determination Program?

A. A TS support system is inoperable and supports two or more TS supported systems

B. A TS supported system is inoperable due to two or more support system inoperabilities

C. An LCO does not allow separate condition entry and a second required system becomes inoperable after the LCO has already been entered

D. A surveillance performed utilizing the Allowed Performance Time of a LCO results in declaring a system required by the LCO inoperable

Proposed Answer: **B**

Applicant References: **None**

Explanation: **The MOST is defined in NDAP-QA-0312 for each TS supported system to ensure supported system LCO Allowed Outage Times (AOTs) are not exceeded due to multiple support system inoperabilities. The MOST is calculated by combining the limiting AOTs for the support system(s) with the limiting AOT for the supported system.**

- **A** Incorrect. No concern for exceeding supported system AOTs exists when only 1 TS support system is inoperable.
- **B** Correct. The instance of 2 or more support system inoperabilities requires tracking MOST for the supported system per NDAP-QA-0312 Step 6.3.5.
- **C** Incorrect. This is a plausible distractor in that it is a description of when application of Completion Time extension is allowed by TS 1.3.
- **D** Incorrect. This describes a condition where some additional action with regard to the AOT is plausible, but it is not related to MOST.

10CFR55: **43.2** This question is SRO-level in that it requires knowledge of generic TS bases to analyze TS required actions (application of MOST).

Technical References: **NDAP-QA-0312**

Learning Objectives: **14635**

Question Source: **Bank** ILO LXR AD044/14620/005

Previous NRC Exam: **No**

Comments

Operations Reviewer MJ / 03JAN14
Init / date

Facility Representative ____/______
Init / date

Exam	SRO	Tier	3	Group	N/A	Cognitive Level	High	Level of Difficulty	2
K/A		2.3.14 Radiation Control				Importance		3.8	
Statement		Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.							

QUESTION 96

Unit 1 is in a refueling outage. Core shuffle is in progress.

A re-channeled irradiated fuel assembly is located in the Fuel Prep Machine at the full-up position for channel fastener installation.

An inadvertent drain path from the reactor to the Suppression Pool is created.

Reactor cavity level lowers rapidly.

The 818' refuel floor is evacuated due to dose rates before the fuel assembly in the prep machine can be lowered.

Initial attempts to secure the leak or makeup to the reactor fail.

Which one of the following describes the initial Emergency Classification for this event, and the basis for the declaration?

	Classification	EAL	Basis
A.	Unusual Event	CU4	Loss or potential loss of the integrity of the Reactor Coolant System fission product barrier represents a potential degradation of the level of safety of the plant
B.	Alert	CA5	Loss of RCS inventory will result in a potential loss of decay heat removal and fuel clad damage
C.	Alert	RA3	Loss of spent fuel pool inventory will result in unexpected increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment
D.	Site Area Emergency	CS5	Loss of RCS inventory will result in a loss or potential loss of two fission product barriers (fuel clad, RCS)

Proposed Answer: C

Applicant References: EP-RM-004 Table R, Table C

Explanation: The event described is an inadvertent loss of RCS and SFP inventory resulting in lowering level in the combined SFP/reactor cavity. EALs from both Table C, for the reactor, and Table R, for the SFP, apply. The event is complicated by the presence of an irradiated fuel assembly in the Fuel Prep Machine which will be uncovered well in advance of the other fuel in the SFP or reactor.

A Incorrect. Alert conditions have been met in RA3. While the coolant inventory loss may have been assumed to exceed the CU4 criteria, this EAL does not apply in Mode 5.

B Incorrect. Reactor level has not lowered to the ECCS initiation setpoint and nothing implying a loss of reactor level indication is specified in the stem.

C Correct. SFP water level will be < 22 ft above the seated irradiated fuel in the SFP and uncovery of the fuel bundle in the prep machine should be assumed as no action to mitigate the draindown has yet been successful.

D Incorrect. CS5 would be the upgrade path in the reactor draindown event continues, but conditions for declaration of this EAL are not yet met as level is not specified and nothing implying a loss of reactor level indication is specified in the stem.

10CFR55	43.4	This is an SRO-level question as an EAL declaration is required to be made.
Technical References	EP-RM-004	
Learning Objectives	14594, 15549	
Question Source	New	
Previous NRC Exam	No	
Comments		

Operations Reviewer MJ / 03JUN14
Init / date

Facility Representative ____/______
Init / date

Exam	SRO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	2
K/A		**2.4.23 Emergency Procedures / Plan**				Importance		**4.4**	
Statement		**Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.**							

QUESTION 97

Unit 2 was operating at rated power when a small reactor coolant system leak developed in the Drywell.

Scram Imminent actions were performed and the reactor was manually scrammed.

Multiple control rods failed to insert. Immediate operator actions for an ATWS were performed.

The following conditions were reported during the scram report:

Reactor level	+25", steady
Reactor pressure	950 psig, steady
Drywell pressure	3 psig, up slow
Suppression Pool temperature	95 °F, up fast

Initial ATWS power was recorded as 35 percent.

Which one of the following identifies the direction to be provided first when implementing EO-200-113 for these conditions?

A. Inject SLC per OP-253-001, SLC System

B. Inhibit ADS per OP-283-001, Automatic Depressurization System and SRVs

C. Lower reactor water level to -60" to -110" per OP-245-005, Infrequent Manual RFP Operations

D. Open SRVs to lower reactor pressure to 945 psig per OP-283-001, Automatic Depressurization System and SRVs

Proposed Answer **A**

Applicant References **None**

Explanation **The stem describes a high-power ATWS in progress. Only immediate operator actions have been performed. EO-200-113 has been performed to the point of recording the initial ATWS power. The four choices represent valid initial directions in each of the power, level and pressure legs for a high-power ATWS. Injection of SLC is the first priority, however, as that will be most effective in reducing power and terminating the ATWS event.**

A Correct. Per EO-200-113 Step LQ/Q-3,

> **If initial ATWS power was greater than 5%, then a relatively large number of control rods have failed to insert. The seriousness of this condition requires immediate injection of boron to positively terminate the ATWS event.**

B Incorrect. Inhibiting ADS is not immediately required as conditions for automatic ADS initiation are not present. Preventing future ADS operation is required, but injection of SLC is the priority per LQ/Q-3.

C Incorrect. While OP-245-005 Att B contains the directives to lower power through tripping recirc pumps and lowering level, the initial goal of the level reduction is to promptly establish conditions to preclude development of severe power/flow instabilities.

D Incorrect. Reactor pressure is being maintained steady by Turbine EHC. Reactor pressure steady implies that cyclic SRV operation is not occurring. No action to lower pressure is therefore required by LQ/P-3.

10CFR55	43.5	This is an SRO-level questions as an assessment of plant conditions is required to identify a high-power ATWS in progress with no mitigating action taken, and selection of the highest-priority action among applicable EOP pathways to implement in response.
Technical References	EO-000-113	
Learning Objectives	14622	
Question Source	New	
Previous NRC Exam	No	
Comments		

Operations Reviewer mj / 05/16/14
Init / date

Facility Representative ____/________
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Exam	SRO	Tier	3	Group	N/A	Cognitive Level	High	Level of Difficulty	3
K/A		2.2.19 Equipment Control				Importance		3.4	
Statement		Knowledge of maintenance work order requirements.							

QUESTION 98

Unit 1 is operating at rated power.

Standby Gas Treatment Fan 0V109B fails during a surveillance test. The fan motor must be replaced.

Which one of the following identifies the appropriate component Criticality Code and Priority of the WO to repair the motor for SBGT Fan B, per NDAP-QA-1901?

A. High Critical
WO Priority 1

B. Critical
WO Priority 2

C. Critical
WO Priority 3

D. Non-Critical
WO Priority 3

Proposed Answer **B**

Applicant References **None**

Explanation **NDAP-QA-1901 Step 5.2 provides the definitions of critical and non-critical components. Component criticality is a key input to NDAP-QA-1901 Att B for properly selecting only those Priority 1 WO that need to bypass the normal scheduling process and may be directed to work around the clock by the Shift Manager. The SRO has the responsibility of determining if maintenance is a Priority 1 condition per Att B.**

A **Incorrect. Loss of a SBGT train does not require an immediate scram, loss of an entire safety system (i.e., safety function) or entry into a LCO shutdown statement. TS 3.6.4.3 allows 7 days for the restoration of the train.**

B **Correct. Loss of the SBGT B fan only renders 1 division of SBGT inoperable and requires entry into a 7-day LCO in TS 3.6.4.3.**

C **Incorrect. Priority 3 WO are associated with operable SSCs or Non-Critical components.**

D **Incorrect. SBGT is critical and should be worked as Pri 2.**

10CFR55 **43.5** **This is an SRO-level question as the knowledge tested is required to correctly determine maintenance WO prioritization and the procedure processes required to be followed to implement the WO.**

Technical References **NDAP-QA-1901 Step 5.2, Att B**
TS 3.6.4.3

Learning Objectives **15268**

Question Source **Modified Bank** **Adapted to SSES from GGNS 2010-06-FINAL**

SUSQUEHANNA STEAM ELECTRIC STATION
LOC26 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Previous NRC Exam Yes

Comments

23

Operations Reviewer mj / 06/~~0~~3/14
Init / date

Facility Representative ____/______
Init / date

Exam	SRO	Tier	3	Group	N/A	Cognitive Level	High	Level of Difficulty	2
K/A		2.4.32 Emergency Procedures / Plan				Importance		4.0	
Statement		Knowledge of operator response to loss of all annunciators.							

QUESTION 99

Use your provided references when answering this question.

Unit 1 is shutdown for a refueling outage in a divisional outage window.

Unit 2 is operating at rated power.

All annunciators on the 1C601 and 2C601 panels and OC653 are lost due to an electrical disturbance.

Unit 1 receives a spurious reactor scram signal due to the loss of power.

Unit 2 experiences a spurious isolation of RWCU.

Technical Specification requirements for OPERABLE electrical distribution systems are met on both units.

Which one of the following describes the initial Emergency Classification for this event?

A. Unusual Event for Unit 2

B. Unusual Event for both units

C. Alert for Unit 1

D. Alert for both units

Proposed Answer: **A**

Applicant References: **EP-RM-004 Table M, C**

Explanation: **A loss of annunciators has occurred for all ECCS systems on both Units 1 and 2. EAL MU5 applies on Unit 2, as this meets the criteria of > 75 percent of annunciators listed on Table M-3 (ECCS, isolation, effluent radiation, electrcal/DG). No EAL is required for units in Modes 4 and 5, so no declaration is required for Unit 1, even though it has experienced a reactor scram, which meets the definition of a signficant transient per Table M-4.**

The specification of TS-required electrical distribution operable precludes a declaration on Unit 1 on loss of AC or DC power.

A Correct. The loss of annuciation meets the criteria of EAL MU5. RWCU isolation does not meet the criteria for a significant transient per Table M-4. No indication of a loss of PICSY or SPDS is provided.

B Incorrect. The event declaration is not applicable to Unit 1 as no EAL related to loss of annunciation applies.

C **Incorrect. The event declaration is not applicable to Unit 1 as no EAL related to loss of annunciation applies.**

D **Incorrect. The event declaration is not applicable to Unit 1 as no EAL related to loss of annunciation applies.**

10CFR55	**43.5**	**This is an SRO-level question as an EAL declaration is required to be made.**
Technical References	**EP-RM-004**	
Learning Objectives	**14594, 15549**	
Question Source	**New**	
Previous NRC Exam	**No**	
Comments		

Operations Reviewer MJ / 06/23/14
Init / date

Facility Representative ____/______
Init / date

Exam	SRO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	2
K/A	2.3.6 Radiation Control					Importance	3.8		
Statement	Ability to approve release permits.								

QUESTION 100

Which one of the following is required to provide the final authorization of all radioactive liquid effluent releases from the plant?

A. Unit 1 Unit Supervisor

B. Field Unit Supervisor

C. Unit Supervisor – Work Control

D. Shift Manager

Proposed Answer: **D**

Applicant References: **None**

Explanation: **NDAP-QA-0310 Step 4.1.1 requires the Shift Manager to provide final authorization of all liquid effluent releases. OP-069-050 allows the FUS to document obtaining SM approval, but does not authorize the FUS to approve release without the Shift Manager's approval.**

A Incorrect. The Unit 1 Unit Supervisor is responsible for all common equipment and is therefore a plausible distractor.

B Incorrect. The FUS provides direction to initiate the process of obtaining a release permit and approves of rad monitor setup and bypassing of interlocks if required. The FUS cannot direct releases to commence without Shift Manager approval.

C Incorrect. The USW is generically involved in Ops shift activities and is therefore a plausible distractor.

D Correct. The Shift Manager is specifically identified as the final authorization to commence liquid effluent releases from the plant by NDAP-QA-0310.

10CFR55: **43.4** **This is an SRO-level question as it relates to the approval process for liquid radwaste release permits.**

Technical References: **NDAP-QA-0310**

Learning Objectives: **15314**

Question Source: **Bank** **ILO LXR AD044/15314/021**

Previous NRC Exam: **No**

Comments

Operations Reviewer MJ / 03Jun 14
Init / date

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