

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

1

ID: RS10003-N06

Points: 1.00

Unit 1 tripped from 100% power.

The cause of the trip was a faulty main generator differential overcurrent relay.

Tripping the RCPs would be required if ...

- A. 1CC685, Thermal Barrier Isolation Valve, was inadvertently closed.
- B. the running 1A CV pump trips and the 1B CV pump cannot be started.
- C. Cnmt pressure transmitter (1PT-936) failed high with a previously failed Cnmt pressure channel (1PT-937) bistables in their required Tech Spec condition.
- D. SSPS train A slave relay K626 (CNMT ISOL PHASE B) inadvertently actuated and cannot be reset.

Answer: D

Answer Explanation

Question: 1

Reference Provided: No

History: Modified for LORT bank question

RO

Cog level: High

K/A 003 K1.01 -Importance RO 2.6 SRO 2.8

003 Reactor Coolant Pump K1.01 Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems: RCP lube oil

Tier: 2

Group: 1

K/A match: This question requires examinee knowledge of physical connection of RCPS to the RCS lube oil system and the effect that a loss cooling of that system will have.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because if CC thermal barrier cooling and seal injection are lost, the RCPs must be tripped to prevent damage to the RCP seals. As long as seal injection flows are maintained, cooling to the RCP thermal barrier is not lost and 1CC685 closing will not by itself require an RCP trip, making this choice incorrect.

Choice B is incorrect. This is plausible because if seal injection and CC thermal barrier cooling are lost, the RCPs must be tripped to prevent damage to the RCP seals. As long as CC thermal barrier cooling flow is maintained, cooling to the thermal barrier is not lost and loss of seal injection flow will not by itself require an RCP trip, making this choice incorrect.

Choice C is incorrect. This is plausible since tripping bistables normally causes actuation of the instruments bistables. This would cause 2 of 4 channels to actuate Phase B and isolate CC cooling to the RCPs. In this case phase B HI-3 CNMT pressure bistables are energize to actuate, and bypassing the bistable per Tech Specs would cause the logic to become 2/3 for actuation. This is why this choice is incorrect.

Choice D is correct. A Phase B isolation isolates CC cooling to the RCP motor oil coolers requiring the RCPs to be tripped.

Task No: R-OA-061

Learn Objective: 4D.OA-51

Reference: BwAR 1-5-A7 rev 6

EXAMINATION ANSWER KEY

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2

ID: RS20003-N01

Points: 1.00

Which of the following conditions will cause the middle indicating light shown below to be LIT?



- A. LOW lube oil pressure WITH the Oil Lift Pump RUNNING.
- B. ADEQUATE lube oil pressure WITH the Oil Lift Pump RUNNING.
- C. Lube Oil Reservoir Level HIGH WITH the Oil Lift Pump SECURE.
- D. Lube Oil Reservoir Level LOW WITH the Oil Lift Pump SECURE.

Answer: B

EXAMINATION ANSWER KEY

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Answer Explanation

Question: 2

Reference Provided: No

History: New for 2018 NRC exam

RO **Cog level:** Memory

K/A003A4.03-Importance RO 2.5 SRO 2.8

003 Reactor Coolant Pump System (RCPS)

A4.03 Ability to manually operate and/or monitor in the control room: RCP lube oil and lift pump motor controls.

Tier: 2

Group: 1

K/A match: The question requires examinee knowledge of interlocks which provide for adequate lubrication of the RCP prior to starting.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because the middle light on breaker control switches is normally a trip or trouble indication. The indication for the RCP lift oil pump is the pressure is adequate for the interlock, making this choice incorrect.

Choice B is correct. An RCP requires 600 psig lift oil pressure as a starting interlock. This interlock being met is indicated by the white light being lit.

Choice C is incorrect. This is plausible because the middle light on pump control switches is normally an interlock or trouble indication. The indication for the RCP lift oil pump is the pressure is adequate for the interlock, making this choice incorrect.

Choice D is incorrect. This is plausible because the middle light on breaker control switches is normally an interlock or trouble indication. The indication for the RCP lift oil pump is the pressure is adequate for the interlock, making this choice incorrect.

Task No: R-RC-006

Learn Objective: 3C.RC-04-H.

Reference: 20E-1-4030RC01 rev V and 20E-1-4030RC05 rev L

EXAMINATION ANSWER KEY

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3

ID: RS20003-N03

Points: 1.00

Unit 1 at 100% power.

- RMCS is unavailable.
- At 1300, VCT level on both channels is 40% and lowering 1% per minute due to an unisolable letdown line leak.

With NO operator action, what is the latest time the VCT level will stop lowering without damaging the CV pumps?

- A. 1303
- B. 1320
- C. 1335
- D. 1340

Answer: C

Answer Explanation

Question: 3

Reference Provided: No

History: New for 2018 NRC exam

RO

Cog level: High

K/A-004K4.12 Importance RO 3.1 SRO

004 Chemical and Volume Control K4.12 Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: minimum level of VCT.

Tier: 2

Group: 1

K/A match: This question requires the examinee to know the level at which the suction will align to the RWST to prevent losing suction to the CV pumps from the VCT.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because the VCT auto makeup would occur at 37% and without makeup capability, the only way to ensure VCT level does not lower more is to align the RWST. This is incorrect because the RWST swap will automatically occur at 5%.

Choice B is incorrect. This is plausible because the VCT low level alarm will actuate at 20%. This would be an indication that make up is required and without makeup capability the only way to ensure VCT level does not lower more is to align the RWST. The automatic swap to the RWST won't occur until 5% which makes this choice incorrect.

Choice C is correct. At 1% per minute, it will take 35 minutes for the VCT level to drop to 5% and swap to the RWST suction. At this time the VCT level will begin to rise.

Choice D is incorrect. This is plausible because the VCT level would indicate 0 at this time. This is the minimum readable level and cannot drop any lower. The automatic swap to the RWST takes place at 5% to protect the CV pumps from gas binding making this answer incorrect.

Task No: R-CV-005

Learn Objective: 3C.CV-12-A

Reference: Big note CV-3 rev 15

EXAMINATION ANSWER KEY

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4

ID: RS10004-N02

Points: 1.00

Unit 1 is in Mode 5.

- Pressurizer is in a solid condition.
- 1CV128, RH TO CV LTDWN FLOW CONT VLV, is open.
- 1A RH train is in standby.
- 1B RH train is in shutdown cooling.

If 1PK-0131, LTDWN LINE PRESSURE CONT VLV 1CV131 controller, SP up arrow pushbutton is depressed for 3 seconds and released, 1CV131 valve will throttle (1) and RCS pressure will (2).



- A. (1) OPEN
(2) LOWER
- B. (1) CLOSED
(2) RISE
- C. (1) OPEN
(2) RISE
- D. (1) CLOSED
(2) LOWER

Answer: B

EXAMINATION ANSWER KEY

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Answer Explanation

Question: 4

Reference Provided: No

History: Bank 2011 NRC exam **RO** **Cog level:** High

K/A005K5.05 Importance RO2.7 SRO 3.1

005 Residual Heat Removal K5.05 Knowledge of the operational implications of the following concepts as they apply to the RHRS: Plant response during "solid plant" pressure change due to the relative incompressibility of water.

Tier: 2

Group: 1

K/A match: The question requires examinee ability to predict changes in RCS pressure associated with "solid plant" conditions and the incompressibility of water when the pressure control valve throttles closed.

10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because 1PK-131 setpoint button being depressed in the upward causes a reverse action in the 1CV131 valve. The 1PK-131 setpoint button being depressed in the ups direction raises the setpoint requiring the 1CV131 valve to close to raise pressure, making this incorrect.

Choice B is correct. The 1PK-131 controller setpoint controls the pressure upstream of the 1CV131 pressure control valve. Increasing the pressure in the entire RCS behind the valve because the RCS is water solid.

Choice C is incorrect. This is plausible because 1PK-131 setpoint button being depressed in the upward causes a reverse action in the 1CV131 valve instead of directly controlling the valve. The controller directly controls RCS pressure during solid plant operations so raising the 1PK-131 setpoint would also raise pressure. The pressure control is a reverse acting controller, making this choice incorrect.

Choice D is incorrect. This is plausible because raising the setpoint will cause the 1CV131 to close. When 1CV131 closes, the letdown pressure downstream of the valve will drop and since the RCS is solid it is plausible that RCS pressure would drop with the downstream pressure. When 1CV131 closes, less water will be coming out of the RCS than going in which will cause pressure to rise, making this choice incorrect.

Task No: R-CV-006

Learn Objective: 3S.CV-01-A

Reference: Big note CV-1 rev. 16 and RH-1 rev.10 I1-CV-XL-01 CVCS lesson plan rev. 9

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

6

ID: RS10007-N03

Points: 1.00

Unit 1 is at 100% power.

- A Pressurizer PORV fails open.
- The associated PZR PORV Block valve will NOT close.
- Operators perform a manual reactor trip and safety injection.
- The NSO has just reported the PRT rupture disk has blown.

In addition to lowering PRT pressure, 5 minutes later, which of the following indications is consistent with the PRT rupture disk opening?

- A. PRT level RISING.
- B. PRT level is LOWERING at 1% per minute.
- C. Containment pressure is RISING at 5 psig per minute.
- D. Containment RF sump leak detection flow rates are RISING.

Answer: D

EXAMINATION ANSWER KEY

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Answer Explanation

Question 6

Reference provided: no

History: Bank (Braidwood NRC Exam 2011 Question #35) RO

Cog level: High

K/A: 007K1.01 **Importance:** RO 2.9 SRO 3.1

007 Pressurizer Relief Tank/Quench Tank System (PRTS)

K1.01 Knowledge of the physical connections and/or cause effect relationship between the PRTS and the following system: Containment

TIER: 2

GROUP: 1

K/A match: This question requires examinee knowledge of effect of PRT rupture on containment.

10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible as lowering PRT pressure would lower back pressure allowing level to rise faster. The PRT level will be rising even if the rupture disk did not open because RCS pressure is much greater than PRT rupture disk pressure and would still flow steam and water to the PRT, making this incorrect.

Choice B is incorrect. This is plausible as the PRT is being filled with RCS at a high temperature and the PRT could flash to steam lowering level. The level drop would be very slight compared to the input from the RCS and level will tend to be trending up or off scale when the rupture disk opens making this incorrect.

Choice C is incorrect. This is plausible because the energy being released from the PZR PORV is now being put into containment. The pressure rise would not be 5 psig per minute based on a single PZR PORV, this rate is more indicative of a large break LOCA, making this incorrect.

Choice D is correct. The sump flow rates would rise as a result of the steam from the PRT and condensation input to cnmt floor drain sump.

Task No: R-RY-013

Obj No: S.RY1-15

Reference(s): ILT LP I1-RY-XL-01 rev 7b, Bwd Big Note RW-3 rev. 0

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

7

ID: RE10026-N03

Points: 1.00

Both units are at 100% power.

- The 0 CC HX and pump are aligned to Unit 2.
- A CC system malfunction has resulted in placing ALL CC pumps aligned to Unit 2 in PTL.

What is the effect on the Unit 1 CC system, if any?

- A. NO effect on Unit 1 CC system.
- B. Loss of CC flow capability to the 1A RH train.
- C. Loss of CC flow capability to the 1B RH train.
- D. Loss of ALL CC flow capability to spent fuel pool cooling.

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 7

Reference Provided: NO

History: Bank Question 2013 NRC exam RO **Cog level:** High

K/A: 008K3.01 **Importance:** RO 3.4 SRO 3.5

008 Component Cooling Water

K3.01 Knowledge of the effect that a loss or malfunction of the CCWS will have on the following:
Loads cooled by CCWS

Tier: 2

Group: 1

K/A match: The examinee must determine the effect of loss of Unit 2 CCW (with the 0 CC HX and pump aligned) on the various CCW flow headers. In order to determine this, examinee must understand how the system is aligned (thus establishing the reason for which CC support is lost).

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because systems AC distribution and SX are connected to both units at the same time and a malfunction on a Unit would normally have no physical impact on the other unit. The only impact would normally be entering an LCO for not having a cross connection ability. The Unit 0 CC heat exchanger provides for A safety train which provide for the A RH heat exchanger cooling for both units, making this choice incorrect.

Choice B is correct. With Unit 0 CC HX and pump aligned to Unit 2, the Unit 2 side supports BOTH units' A train safety loops (A RH trains).

Choice C is incorrect. This is plausible because the CC system is broken into safety loops and the common HX provides for a safety loop and on the big note, the B trains are next to each other. This would be easily confused and seem as though they are connected to the common HX. The common HX provides the A train safety loop, making this choice incorrect.

Choice D is incorrect. Because the SFP is a common system between units, and the 0 CC HX and pump are aligned to Unit 2, an examinee may well assume that all cooling is lost to SFP cooling. SFP cooling has separate Unit 1 and Unit 2 side HXs supplied by their non-safety flow loops.

Task No: R-CC-002

Learn Objective: 4C.CC-02

Reference: Big Note CC-1, rev. 17, CC system. Lesson Plan I1-CC-XL-01 rev. 5a.

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

8

ID: RS20008-N04

Points: 1.00

Unit 1 Reactor Trip and SI occurred due to a faulted SG inside containment.

- Phase B actuation has occurred.
- RCS temperature is currently 345°F and stable.
- RCS pressure is currently 350 psig and stable.
- The faulted SG is isolated and has completely blown down.
- SI termination is complete.

What is the available mode of RCS heat removal?

- A. Forced circulation with 1 RCP.
- B. Forced circulation with ALL RCPs running.
- C. Natural circulation with the steam dumps.
- D. Placing an RH train in service.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 8

Reference Provided: No

History: New question for 2018 NRC exam

RO **Cog level:** High

K/A-008 G2.4.21 Importance RO 4.0 SRO 4.6

008 Component Cooling Water G2.4.21 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.

Tier: 2

Group: 1

K/A match: The question requires the examinee to determine from the given parameters the preferred heat removal source is RH heat removal, which uses the CC system as its heat sink.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because 1BwEP ES-1.1 SI Termination directs attempting to start RCPs for pressure control. Since all the RCP motors have been wetted by CS and steam they will not be started unless needed for core cooling in an emergency, making this choice incorrect.

Choice B is incorrect. This is plausible because forced circulation cooldown ensures proper mixing of the RCS and prevents pockets of diluted water from hiding out and being introduced to the core. Since all the RCP motors have been wetted by CS and steam, they will not be started unless needed in an emergency, making this choice incorrect.

Choice C is incorrect. This is plausible since offsite power is not lost, it is preferred to steam the SG to the condenser to preserve CST inventory and minimize impact on surrounding areas by using steam via the SG PORVs. Phase B actuation implies that containment pressure rose greater than the MSIV closure pressure of 8.2 psig making the steam dumps unavailable.

Choice D is correct. With RCS pressure below 360 psig and RCS temperature below 350F, the RH system can be aligned for shutdown cooling. This will allow the SG PORVs to be closed eventually and conserve CST inventory and minimize the impact of secondary steam from being released to the environment.

Task No: R-RH-001

Learn Objective: 4C.RH-03

Reference: BwOP RH-6 rev 57

EXAMINATION ANSWER KEY

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9

Points: 1.00

Unit 1 is in Mode 3 NOP/NOT.

- The crew is performing 1BwOSR 3.4.11.2, PRESSURIZER SYSTEM PORV VALVE STROKE SURVEILLANCE.
- The NSO is monitoring PRT pressure and temperature on the 1PM05J meters.

During the PZR PORV surveillance, expected PRT pressure will (1) and expected PRT temperature will (2).

- A. (1) rise
(2) rise
- B. (1) be unchanged
(2) rise
- C. (1) rise
(2) be unchanged
- D. (1) be unchanged
(2) be unchanged

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 9

Reference Provided: No

History: new for 2018 NRC exam

RO

Cog level: High

K/A-010 A3.01 Importance RO 3.0 SRO 3.2

010 Pressurizer Pressure Control System A3.01 Ability to monitor automatic operation of the PZR PCS, including PRT temperature and pressure during PORV testing.

Tier: 2

Group: 1

K/A match: The question requires the examinee to understand how the PORV testing affects the PRT.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. It is plausible, PORV isolation valves have leaked by in the past and this would cause PRT pressure and temperature to rise. No indication of leak by was given in the stem.

Choice B is incorrect. It is plausible that temperature would rise from steam trapped between the PORV and the PORV isolation valve causing the water in the PRT to absorb the heat of condensation and raise the PRT temperature without significant pressure rise. The volume of steam trapped would not be significant enough to raise PRT temperature.

Choice C is incorrect. It is plausible that hot water trapped between the PORV and the isolation valve at high pressure would be released into the PRT and flash to steam causing a pressure rise but not be a significant enough mass to affect temperature. The volume of water trapped would not be significant enough to raise PRT pressure.

Choice D is correct. The pressurizer PORVs are isolated by their block valves before testing and the PRT sees no energy addition during PORV testing.

Task No: R-RY-001

Learn Objective: 3C.RY-02-A

Reference: 1BwOSR 3.4.11.2 rev 006

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

10

ID: RS20010-N01

Points: 1.00

Unit 1 is in Mode 5.

- BwOP RC-19, VACUUM FILLING AN ISOLATED REACTOR COOLANT LOOP/PRESSURIZER, is in progress.
- Pressurizer level at 70% cold cal.
- Pressurizer pressure at 10" Hg absolute.
- Vacuum pump is shut down and isolated.
- RH is in service with RCS loop temperatures at 115°F.
- Pressurizer heaters are energized.

Assuming pressurizer pressure remains constant, which of the following most closely approximates pressurizer liquid temperature when the bubble STARTS forming in the pressurizer?

- A. 115°F
- B. 162°F
- C. 212°F
- D. 228°F

Answer: B

Answer Explanation
<p>Question: 10</p> <p>Reference Provided: Steam Tables</p> <p>History: Braidwood NRC Exam 2009 RO Cog level: High</p> <p>K/A-010K5.01 Importance RO 3.5 SRO 4.0</p> <p>010 Pressurizer Pressure Control System K5.01 Knowledge of the operational implications of the following concepts as they apply to the PZR PCS: Determination of condition of fluid in PZR, using steam tables.</p> <p>Tier: 2</p> <p>Group: 1</p> <p>K/A match: The question meets the K/A, requires examinee knowledge of steam table use in determining saturation temperature in pressurizer.</p> <p>10CFR55.41(b)(14) Principles of heat transfer thermodynamics and fluid mechanics.</p> <p>Answer explanation and plausibility:</p> <p>Choice A is incorrect. This is plausible because this is the current RCS temp and, therefore, PZR temp is also 115°F. So, drawing a vacuum to begin forming a bubble could imply lowering pressure to saturated conditions for 115°F.</p> <p>Choice B is correct. 10"Hg absolute is approx. 5 psia which has a saturation temp. of 162°F.</p> <p>Choice C is incorrect. This is plausible because sat. temp for atmospheric pressure 212°F.</p> <p>Choice D is incorrect. This is plausible because sat. temp for 20 psia or 5 psig. (common error for converting Hg to psia 228°F.)</p> <p>Task No: R-RC-038</p> <p>Learn Objective: 3C.RC-14</p> <p>Reference: BwOP RC-19 rev 24</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

11

ID: RE10007-N02

Points: 1.00

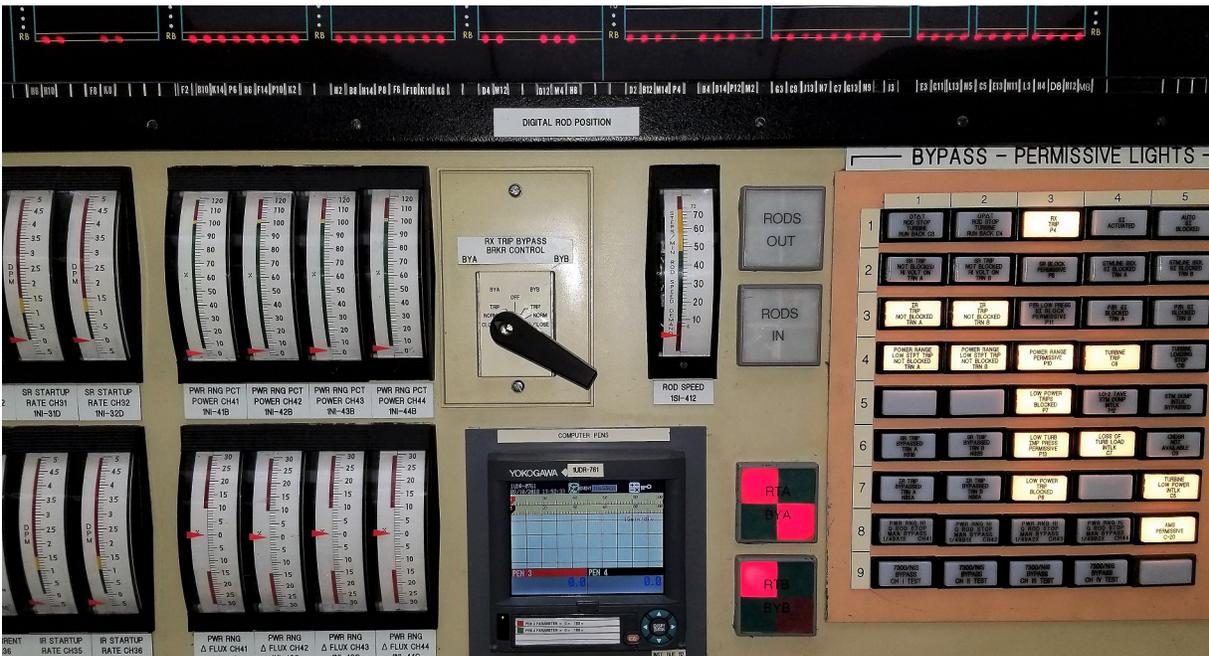
Unit 1 was at 100% power.

- 1BwOSR 3.3.1.4-1, SSPS, REACTOR TRIP BREAKER, AND REACTOR TRIP BYPASS BREAKER SURVEILLANCE (TRAIN A), testing was in progress.

An event occurs and ONLY the following actions have been taken.

- Alarms and Bypass Permissive Panel have been acknowledged.
- Bypass Permissive Status Light "P4 RX TRIP" is **FAST FLASHING**.

Current conditions are pictured below.



The NEXT procedurally required action is to...

- initiate emergency boration.
- verify ALL turbine throttle and governor valves CLOSED.
- manually trip the reactor from both the 1PM05J and 1PM06J.
- go to 1BwFR-S.1, "RESPONSE TO NUCLEAR POWER GENERATION/ATWS."

Answer: C

EXAMINATION ANSWER KEY

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Answer Explanation

Question: 11

Reference Provided: No **NOTE:** print in color for exam use.

History: Bank 2013 NRC exam **RO** **Cog level:** High

K/A- 012 A4.01 **Importance** RO 4.5 SRO 4.5

012 Reactor Protection A4.01 Ability to manually operate and/or monitor in the control room: manual trip button.

Tier: 1

Group: 1

K/A match: Meets K/A, examinee must determine that a manual trip is required from the status of the reactor trip and bypass breakers.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect and plausible, emergency boration would be required if the reactor did not completely shut down after attempting to trip the reactor from both main control board switches.

Choice B is incorrect and plausible, verifying turbine throttle and governor valves are closed is an immediate operator action completed after attempting to trip the reactor from both main control board switches.

Choice C is correct. Immediate action step one, if all reactor trip and bypass breakers are not open then manually trip the reactor at 1PM05J and 1PM06J. Step 1 of 1BwEP-0 requires that ALL conditions (rod bottom lights lit, reactor trip AND bypass breakers are OPEN, and neutron flux dropping). Although indications are that the reactor is tripped, the bypass breaker A indicates closed and the response not obtained procedural action requires manually tripping the reactor from BOTH 1PM05J and 1PM06J.

Choice D is incorrect and plausible, this would be the next step to respond to continued power production if all rod bottom lights were not lit.

Task No: R-EP-011

Learn Objective: T.EP01-06A.

Reference: 1BwEP-0 rev 302

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

12

ID: BWLI-DC1-016

Points: 1.00

Unit 1 is at 50% power, normal plant line-up.

A LOSS of DC bus 111 occurs.

With NO operator action, what is the expected plant response?

- A. Reactor trip from LO-2 SG level.
- B. LOSS of field flash for the 1B Diesel Generator.
- C. Reactor trip from HIGH PZR level.
- D. LOSS of power to Bus 159 following the Main Generator trip.

Answer: A

Answer Explanation
<p>Question: 12 Reference Provided: No History: ILT Bank question, no record of previous use. RO Cog level: High K/A-012K1.02 Importance RO 3.4 SRO 3.7 012 Reactor Protection System K1.02 Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems: 125V dc system. Tier: 2 Group: 1 K/A match: The question requires knowledge of the effect a loss of DC will have on the plant and the RPS response. 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.</p> <p>Answer explanation and plausibility: Choice A is correct. The loss of DC bus 111 causes many FW isolation valves to close, the loss of feedwater lowers SG NR level to the RPS setpoint and the reactor trips. Choice B is incorrect. This is plausible because field flash is provided by the respective train of DC power to each diesel generator. This is incorrect because the 1B diesel generator is provided field flash from DC bus 112. Choice C is incorrect. This is plausible because letdown isolation will occur on loss of DC 111, this causes pressurizer level to rise on charging flow alone. This incorrect because charging flow will automatically reduce to minimum and take much longer to reach the trip setpoint than SG level. Choice D is incorrect. This is plausible because bus 159 control power is from DC bus 111, and if Bus 159 was supplied by the UAT, it would not be able to automatically transfer to the SAT. This is incorrect because Bus 159 is normally aligned to the SAT and would not need to be swapped after the DC 111 loss of power.</p> <p>Task No: R-OA-007 Learn Objective: 4D.OA-23 Reference: 1BwOA ELEC-1 rev 111</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

13

ID: RS20013-N03

Points: 1.00

Unit 1 is at 100% power.

- Containment pressure transmitter 1PT-935 failed HIGH.
- The SM has directed the 1PT-935 bistables be placed in bypass.

When 1PT-935 bistables are placed in bypass, the coincidence logic for Containment Spray Actuation will change from (1) with 1PT-935 failed HIGH to (2) when 1PT-935 bistables are bypassed.

- A. (1) 1 of 3
(2) 2 of 3
- B. (1) 2 of 3
(2) 1 of 3
- C. (1) 2 of 3
(2) 2 of 3
- D. (1) 1 of 3
(2) 1 of 3

Answer: A

EXAMINATION ANSWER KEY

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Answer Explanation

Question: 13

Reference Provided: No

History: New for 2018 NRC exam RO **Cog level:** High

K/A-013 K6.01 Importance RO 2.7 SRO 3.1

013 Engineered Safety Features Actuation System (ESFAS) K6.01 Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors.

Tier: 2

Group: 1

K/A match: The question requires the examinee to understand how the loss of a CS actuation sensors (instrument) and subsequent actions will impact the actuation circuitry.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Answer explanation and plausibility:

Choice A is correct. 2 of 4 channels of containment pressure are required to actuate CS. With the one channel failed and actuated, only 1 of the remaining 3 channels is needed. By placing the channel in bypass, the failed channel will not actuate so 2 of the remaining 3 channels are required to actuate CS.

Choice B is incorrect, 1 of 3 channels are required when the channel is tripped and 2 of 3 when the channel is bypassed. CS actuation is an energize to actuate logic which is backwards from other ESF actuation logics and causes confusion as to whether or not logics are the same for bypassing and tripping, making this a plausible distractor.

Choice C is incorrect, 1 of 3 channels are required when the channel is tripped and 2 of 3 when the channel is bypassed. CS actuation is an energize to actuate logic which is backwards from other ESF actuation logics and causes confusion as to whether or not logics are the same for bypassing and tripping, making this a plausible distractor.

Choice D is incorrect, 1 of 3 channels are required when the channel is tripped and 2 of 3 when the channel is bypassed. CS actuation is an energize to actuate logic which is backwards from other ESF actuation logics and causes confusion as to whether or not logics are the same for bypassing and tripping, making this a plausible distractor.

Task No: R-EF-013

Learn Objective: 3C.EF-01-A-2

Reference: Tech Spec 3.3.2, I1-EF-XL-01 Engineered Safety Features rev 5

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

14

ID: RS10022-N02

Points: 1.00

Unit 1 is at 100% power.

- Annunciator 1-21-B10, BUS 131X CONT PWR FAILURE, alarmed.
- NO other annunciators are in alarm.

Unit 1 safety injection occurs.

The 1A train of RCFCs WILL (1) in LOW speed, and to clear the alarm, the crew will (2).

- A. (1) NOT start
(2) replace the blown control power fuses
- B. (1) NOT start
(2) swap to the alternate control power circuit
- C. (1) start automatically
(2) swap to the alternate control power circuit
- D. (1) start automatically
(2) replace the blown control power fuses

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 14

Reference Provided: No

History: Modified 2011 NRC exam

RO

Cog level: Memory

K/A- 022A2.03 **Importance** RO 2.6 SRO 3.0

022 Containment Cooling System A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Fan motor thermal overload/high-speed operation

Tier: 2

Group: 1

K/A match: This question requires the examinee to predict the impact of the RCFC running in high speed with a loss control power and the effect on the low speed RCFC SI actuation and use the BwAR to correct the problem.

10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible since the low speed fan will not start and control power to some MCCs are fused. Safety related busses do not have fused control power; they have an independent alternate power supply to align, making this choice incorrect.

Choice B is correct. The low speed RCFC breakers will not close on a safety injection and per the BwAR, the control power will be aligned to the alternate power supply per BwOP DC-6.

Choice C is incorrect. This is plausible since safety related systems receive redundant signals to cause actuation, and the swap to alternate power supply is the BwAR directed action. This is incorrect because the redundant actuations signal still must go through a breaker that does not have control power.

Choice D is incorrect. This is plausible since safety related systems receive redundant signals to cause actuation, and control power to some MCCs are fused. This is incorrect because the redundant actuation signal still must go through a breaker that does not have control power and safety related busses do not have fused control power.

Task No: R-VP-002

Learn Objective: 3C.VP-06-C

Reference: 20E-1-4010B rev J, BwAR 1-21-B10 rev 5, BwOP DC-6A1 rev 57

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

15

ID: RS20026-N04

Points: 1.00

Unit 1 had a Reactor trip and SI from full power alignment.

- 1BwEP-0, REACTOR TRIP AND SAFETY INJECTION, is in progress at step 14, CHECK IF CS IS REQUIRED.
- There are NO CS pumps running.

What is the MINIMUM Containment pressure that would require the crew to manually actuate CS?

- A. 4 psig
- B. 6 psig
- C. 16 psig
- D. 22 psig

Answer: D

Answer Explanation

Question: 15

Reference Provided: No

History: New for 2018 NRC exam

RO Cog level: Memory

K/A-026A1.01 Importance RO 3.9 SRO 4.1

026 Containment Spray A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment pressure

Tier: 2

Group: 1

K/A match: Requires the examinee to predict the pressure at which CSS would have to be manually actuated to prevent exceeding containment design limits.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because the pressure is above the SI actuation setpoint. This is incorrect because SI does not automatically start CS.

Choice B is incorrect. This is plausible because the pressure is above the point at which containment is declared adverse. This could be considered part of containment design pressure. This is incorrect because CS is not required to actuate until higher pressure.

Choice C is incorrect. This pressure is above the required pressure for terminating CS and could be seen as the pressure at which you would actuate CS manually, making it plausible. CS is not required to actuate until pressure is above 20 psig, making this distractor incorrect.

Choice D is correct. Containment spray actuates at 20 psig and any pressure above that with no CS pumps running would require operator action to prevent a challenge in containment integrity.

Task No: R-CS-001

Learn Objective: 3D.CS-01-B

Reference: 1BwEP-0 rev 302

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

16

Points: 1.00

Unit 1 is at 100% power.

- A recent RCS leakrate surveillance indicated .015 GPM rise in RCS leakage.
- The US has directed the RO to monitor MCR indications for a possible SG tube leak.

The FIRST alarm indicating a small SG tube leak will be from the...

- A. 1PM04J, "FEEDWATER."
- B. 1PM05J, "ENG SAFETY FEATURES."
- C. 0PR16J, "B/D AFTER FILTER 0A OUTLET."
- D. 1PR27J, "SJAE GS EXH."

Answer: D

Answer Explanation

Question: 16

Reference Provided: No

History: New for 2018 NRC exam RO Cog level: Memory

K/A- 039A1.10 -Importance RO 2.9 SRO 3.0

039 Main and Reheat Steam System A1.10 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Air ejector PRM

Tier: 2

Group: 1

K/A match: This question requires the examinee to understand the impact of a steam generator tube leak on the Air Ejector process rad monitor (1PR27J).

10CFR55.41(b)(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible since the leak flow is from the primary into the secondary through the SG tubes. The rise in input into the SG could cause level or control alarms in the SGWLC system. This is likely in a large SG tube rupture, the leak in the is question is small and will not alarm, making this choice incorrect.

Choice B is incorrect. This is plausible since RCS leakage is identified by VCT level and pressure and PZR level lowering and alarms associated with those indications on 1PM05J. These indications will be obvious in a larger SG tube leak; however, for the small leak in the question, it is likely that these indications will not even change a noticeable amount and will not alarm, making this choice incorrect.

Choice C is incorrect. The SG blowdown RAD monitor 0PR16J will alarm on SG tube leakage and redirect flow to the BD monitor tank making this a plausible distractor. The blowdown after filter RAD alarm is delayed in seeing the leakage because it must cycle through the steam system back through feed and mix with the SG water then pass through the demins before it is sent by the after filter alarm. This will not be the first alarm on the radiation monitoring system.

Choice D is correct. The air ejector RAD monitor is designed to detect small leakage amounts in the SG tubes. This is the first alarm indication of a SG tube leak.

Task No: R-OA-003

Learn Objective: 4D.OA-17, S.BZ1-10

Reference: I1-OA-XL-43 rev 12a, I1-BZ-XL-01 rev3b

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

17

Points: 1.00

Unit 1 was manually tripped from 100% power.

- Immediately after the reactor trip, a loss of DC bus 111 occurred.
- Subsequently, a Main Steam Isolation signal actuated.
- The RO was directed to verify ALL MSIVs and MSIV Bypass valves are closed.
- The RO checked the GROUP 6 MLB-1 MS ISOL status lights.

The GROUP 6 MLB-1 MS ISOL status lights are...

- A. reliable for CLOSED indication ONLY for the 1A and 1D MSIVs and BYPASS valves. The 1B and 1C MSIV and BYPASS valves positions need to be locally verified.
- B. reliable for CLOSED indication ONLY for the 1B and 1C MSIVs and BYPASS valves. The 1A and 1D MSIV and BYPASS valves positions need to be locally verified.
- C. NOT reliable for CLOSED indication for ALL MSIVs AND BYPASS valves. ALL MSIVs AND BYPASS valves positions will need to be locally verified.
- D. reliable for CLOSED indication for ALL MSIVs and BYPASS Valves. NO valves require local verification.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 17

Reference Provided: No

History: New for 2018 NRC exam RO Cog level: High

K/A- 039A3.02 **Importance** RO 3.1 SRO 3.5

039 Main and Reheat Steam Supply System A3.02: Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS

Tier: 2

Group: 1

K/A match: The question requires the examinee to understand the automatic isolation of mainsteam lines and the power supplies which provide for isolation and indication.

10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Answer explanation and plausibility:

DC bus 111 provides active train power for MSIV 1A and 1D and standby train power for MSIVs 1B and 1C.

Choice A is incorrect. This is incorrect because group 6 lights are energized from 2 redundant power supplies and rely on contacts to makeup to indicate position. This is plausible because the MSIVs have redundant actuation power and the bypass valves can only be opened from their primary power; this is a common misunderstanding.

Choice B is incorrect. This is incorrect because group 6 lights are energized from 2 redundant power supplies and rely on contacts to makeup to indicate position. This is plausible because the MSIVs have redundant actuation power and the bypass valves can only be opened from their primary power; this is a common misunderstanding.

Choice C is incorrect. This is incorrect because group 6 lights are energized for 2 redundant power supplies and rely on contacts to makeup to indicate position. This is plausible because the MSIV and bypass valves have a normal power supply for opening and a redundant supply to ensure closure is complete when required.

Choice D is correct. All indications on the group 6 monitor lights will have power available and read position from limit switch contacts based on physical position.

Task No: R-MS-003

Learn Objective: 4C.MS-10

Reference: 20E-1-4030MS001-04 rev r,r,r,s

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

18

ID: RE20059-N02

Points: 1.00

Unit 1 was at 100% power prior to a transient.

1BwEP-0 step 4, CHECK SI STATUS, is in progress.

- 1B SG NR level lowered rapidly to 0%.
- 1B SG pressure is 700 psig and lowering slowly.
- 1A/1C/1D SG NR levels are 21% and rising.
- 1A/1C/1D SG pressures are 900 psig and stable.
- AF is operating as required.
- Containment pressure is 9 psig.
- All Tcold are 537°F and slowly lowering.
- RCS pressure is 2000 psig and slowly lowering.

1. What malfunction has occurred?
2. What is the procedure flow path after 1BwEP-0?
 - A. (1) Steam line break on 1B SG.
(2) 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT then 1BwEP-2, FAULTED SG ISOLATION.
 - B. (1) Feed line break on 1B SG.
(2) 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT then 1BwEP-2, FAULTED SG ISOLATION.
 - C. (1) Steam line break on 1B SG.
(2) 1BwEP-2, FAULTED SG ISOLATION then 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.
 - D. (1) Feed line break on 1B SG.
(2) 1BwEP-2, FAULTED SG ISOLATION then 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 18

Reference Provided: No

History: New question for 2018 NRC exam

RO

Cog level: High

K/A-059 A2.05 Importance RO 3.1 SRO 3.4

059 Main Feedwater A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture in MFW suction or discharge line

Tier: 2

Group: 1

K/A match: The question requires the examinee to predict the impact of a MFW discharge line break and compare it to the indications provided and determine the appropriate procedure to mitigate the consequences.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect, 1BwEP-1 will not be the procedure to mitigate the transient, based on the lowering RCS temp and SG level 1BwEP-0 will direct using 1BwEP-2 . This is plausible because 1BwEP-1 has action to diagnose a secondary break and return to 1BwEP-2.

Choice B is incorrect, 1BwEP-1 will not be the procedure to mitigate the transient, based on the lowering RCS temp and SG level 1BwEP-0 will direct using 1BwEP-2 . This is plausible because a feed line fault is indicated by these conditions and 1BwEP-1 does have steps to diagnose a secondary break and return to 1BwEP-2.

Choice C is incorrect. This is incorrect because a steam line break will cause the effected SG pressure and level to lower more rapidly than the others. This is plausible because 1BwEP-2 will be the mitigating procedure and the steam line break will cause RCS temperature to lower.

Choice D is correct. A feedline break inside containment will cause RCS temperature to lower due to the SG water flashing to steam cooling the RCS. This effect also causes the pressure in that SG to stay elevated until the feed ring is uncovered. Once the steam generator is isolated and blown down the next procedure transition will be to 1BwEP-1 to ensure no other issue exists before transitioning to SI termination.

Task No: R-EP-013

Learn Objective: 3D.EP-06-C

Reference: I1-EP-XL-03

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

19

ID: BWLC3DOA3009

Points: 1.00

Unit 1 was at 100% power.

- DC bus 112 was deenergized.
- The reactor was manually tripped.

The 1B AF pump can be started ...

- A. locally in the 1B AF pump room ONLY.
- B. manually from the MCR or at the RSDP ONLY.
- C. automatically from AMS.
- D. automatically or manually from the MCR.

Answer: A

Answer Explanation

Question: 19

Reference Provided: No

History: LORT bank question RO **Cog level:** Memory

K/A-061 K2.03 Importance RO 4.0 SRO 3.8

061Auxiliary/Emergency Feedwater K2.03 Knowledge of bus power supplies to the following:
AFW diesel driven pump

Tier: 2

Group: 1

K/A match: The examinee must understand the link between DC bus 112 and the diesel driven aux feed pump.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is correct. DC bus 112 provides the control power for the starting circuitry for the remote shutdown panel and MCR. The 1B AF pump has its own starting battery and control power which can be used to start the 1B AF pump from the local control panel.

Choice B is incorrect. This is incorrect because the control signals from SSPS, MCR and RSDP rely on DC bus 112 to provide the starting signal to the local panel. This is plausible since DC bus 112 causes a loss of control power to the train B SSPS equipment which is different from the other control switches in the MCR and the RSDP.

Choice C is incorrect. This is incorrect because the automatic actuation circuitry requires DC bus 112 to start the 1B AF pump from ATWS as well as any other automatic logic. This is plausible as the ATWS system is designed to deliver a redundant start signal to the AF system in the event of a SSPS issue to mitigate a design basis ATWS with a loss of feedwater.

Choice D is incorrect. This is incorrect because the control signals from SSPS, MCR and RSDP rely on DC bus 112 to provide the starting signal to the local panel. This is plausible because the 1B AF pump has an independent battery system to provide starting power and local control power for the starting motor.

Task No: R-AF-005

Learn Objective: 3C.AF-01-B

Reference: 20E-1-4030AF02 rev AC, 20E-1-4030AF12 re AE

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

20

ID: RS10062-C01

Points: 1.00

Unit 2 is at 100% power.

BwOP IP-2, TRANSFERRING AN INSTRUMENT BUS FROM THE INVERTER TO THE CONSTANT VOLTAGE TRANSFORMER, is in progress.

- Operators are in the process of transferring the Instrument Bus 214 power supply to the CVT.

The following are actions/indications at the 214 inverter, 2IP08E:

- The BYPASS SOURCE TO LOAD pushbutton has been depressed.
- The BYPASS SOURCE SUPPLYING LOAD light is LIT.
- Breaker B2, Inverter Output To Static Switch, has been opened.

The operator is about to place the Manual Bypass Switch in BYPASS TO LOAD position.

What effect, if any, will manipulating the Manual Bypass Switch have on MCR indications?

- A. No effect.
- B. The 1AF005H, S/G 1D FLOW CNTRL, will fail closed.
- C. Pressurizer pressure meter, 2PI-458, will momentarily fail to 1700 psig.
- D. Annunciator 2-4-B3, SOLID STATE PROT CAB GENERAL WARNING, will alarm.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 20

Reference Provided: No

History: Previous ILT cert exam (2014) RO **Cog level:** Memory

K/A-062A1.03 Importance RO 2.5 SRO 2.8

SYS 062 AC Electrical Distribution A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AC distribution system controls including: Effect on instrumentation and controls of switching power supplies

Tier: 2

Group: 1

K/A match: The question requires examinee to predict changes in parameters and determine effect on instrumentation in MCR when switching power supplies to instrumentation.

10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Answer explanation and plausibility:

Choice A is correct. Swapping instrument bus power on inverters is a make before break contact. It is a bumpless transfer that will not affect MCR indications per BwOP IP-2 step F.8.A note.

Choice B is incorrect. This is incorrect because the swap is a make before break and instrument power to the 1AF-005H would not be lost. This is plausible because instrument bus 114 provides control power for the 1AF005H and 1BwOA ELEC-2 will have the operator fail open the 1AF005E-H (including H) on a loss of instrument bus 114.

Choice C is incorrect. This is incorrect because pressurizer pressure channel is not affected due to the make before break swap and the 7300 cabinet has a backup power supply that would prevent any loss of indication. This is plausible since 2PI-458 is associated with instrument bus 214 because it is channel 4 of the pressurizer pressure channels (2PA08J).

Choice D is incorrect. This is incorrect because the swap is a make before break and instrument power to the SSPS system would not be lost. This is plausible because if the instrument bus actually lost power, an SSPS general warning would occur.

Task No: R-OA-006

Learn Objective: T.OA02-09

Reference: BwOP IP-2 rev 39

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

21

ID: RS20063-N05

Points: 1.00

Both Units are at 100% power.

- Unit 1 has indication of a DC Bus 111 ground.
- Troubleshooting has isolated the ground to the DC Bus 111 Battery Charger.
- The crew has just crosstied DC Bus 111 to 211 per BwOP DC-7-111, 125V DC ESF BUS 111 CROSS-TIE/RESTORATION and ISOLATED Battery Charger 111.

A Tech Spec entry for 3.8.4 DC Sources-Operating is required IMMEDIATELY for...

- A. BOTH Units.
- B. Unit 1 and will ONLY be required for Unit 2 if completion times are exceeded.
- C. Unit 1 ONLY; NOT required for Unit 2.
- D. Unit 1 and for Unit 2 AFTER an additional 2 hours.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 21

Reference Provided: No

History: New question for 2018 NRC exam

RO

Cog level: Memory

K/A- 063A2.01 **Importance** RO 2.5 SRO 3.2

063 DC electrical distribution A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds

Tier: 2

Group: 1

K/A match: The question requires the examinee to predict the impact of the ground on Unit 1 and the Tech Spec implications for both Units from the procedurally required action of cross tying the DC busses.

10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Answer explanation and plausibility:

Choice A is correct. Tech Spec 3.8.4 is not met on both units, unit 1 because the battery charger is required to meet the LCO, unit 2 because the DC bus is required to be independent of the opposite unit to meet the LCO.

Choice B is incorrect. This is incorrect because Tech Spec 3.8.4 is not met for both units. This is plausible because Unit 1 is the unit with the inoperability and the LCO allows 2 hours to cross tie to the other unit's battery charger.

Choice C is incorrect. This is incorrect because Tech Spec 3.8.4 entry is required both units. This is plausible because the inoperable component is on unit 1 and other Tech Spec functions, such as SX, supply both units and only 1 unit enters Tech Spec for their inoperable pump. It can incorrectly be viewed as the same for the battery charger supply to the DC bus.

Choice D is incorrect. This is incorrect because Tech Spec 3.8.4 entry is required for both units. The inoperability of the DC bus on unit 2 could be delayed by 2 hours if condition A was applied to the bus instead of the specific cross tie condition. This would make this choice plausible.

Task No: R-DC-002

Learn Objective: 4D.DC-01

Reference: Tech Spec 3.8.4, Bases TS 3.8.4

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

22

ID: RS20063-N06

Points: 1.00

Unit 1 was at 100% power.

- The 1A DG #1 Air Receiver relief valve has stuck open.
- The receiver has been isolated and blown down.

The 1A DG starting air system now has enough receiver capacity for (1) DG starts and a Tech Spec entry for the 1A DG (2) required.

- A. (1) 4
(2) IS
- B. (1) 4
(2) IS NOT
- C. (1) 8
(2) IS
- D. (1) 8
(2) IS NOT

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 22

Reference Provided: No

History: New for the 2018 NRC exam RO **Cog level:** Memory

K/A-064K6.07 Importance RO 2.7 SRO 2.9

064 Emergency Diesel Generator K6.07 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers

Tier: 2

Group: 1

K/A match: The question requires the examinee to understand the effect the loss of an air receiver has on the emergency diesel generators.

10CFR55.41(b)(8) Components, capacity, and functions of emergency systems.

Answer explanation and plausibility:

Choice A is incorrect. The EDG air receivers are capable of 4 attempts at starting the EDG and only 1 air receivers is required for the operability of the EDG. This is plausible since most systems require their support systems to remain operable, losing 1 of the 2 air receivers could be incorrectly seen as requiring Tech Spec entry.

Choice B is correct. Only 1 receiver is required for the EDG to be considered operable and the air receiver is capable of 4 starts of the EDG.

Choice C is incorrect. This is incorrect because the total capacity of a single air receiver will NOT support more than 4 starts and entry into Tech Spec is not required for a single receiver out of service. This is plausible since both receivers in operation will support up to 8 start attempts and entry into the Tech Spec is required. Since most systems require their support systems to remain operable, losing 1 of the 2 air receivers could be incorrectly seen as requiring Tech Spec entry.

Choice D is incorrect. This is incorrect because the total capacity of a single air receiver will not support more than 4 starts. This is plausible since both receivers in operation will support up to 8 start attempts and because the EDG support equipment can have 1 air receiver out of service and still support the operability of the EDG.

Task No: R-DG-015

Learn Objective: 3C.DG-01-D

Reference: BwOP DG-1 rev 29

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

23

ID: RS10073-N01

Points: 1.00

The Turbine Bldg Fire and Oil (F & O) sump is transferring water to the west lagoon due to unusually heavy sump input.

- The 0D (low flow) sump pump and 0A (high flow) sump pump are currently running.
- An alarm on the RMS indicates 0PR05J, TB FIRE OIL SUMP, has turned RED.

As a result, the F & O sump system will...

- A. trip ONLY the high flow F & O sump pump and RE-DIRECT the flowpath to the waste water treatment oil separator.
- B. NOT trip the F & O sump pumps, but will RE-DIRECT the flowpath to the turbine bldg floor drain tanks.
- C. trip BOTH F & O sump pumps and TRIP ALL the tendon tunnel sump pumps.
- D. trip BOTH F & O sump pumps and ISOLATE the flowpath to the lagoon.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 23

Reference Provided: No

History: Bank 2011 NRC exam RO **Cog level:** Memory

K/A- 073K4.01 Importance RO 4.0 SRO 4.3

073 Process Radiation Monitoring K4.01: Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint.

Tier: 2

Group: 1

K/A match: This question requires examinee knowledge of interlocks that will isolate releases to the environment upon a high rad signal.

10CFR55.41(b)(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect because the flowpath is isolated and all pumps are tripped. This is plausible because other systems, like SG blowdown, divert flow path on high rads. The waste water treatment flow path is NOT capable of handling flow from the high flow F&O sump pump, so it would have to be tripped if flow was directed to the waste water treatment facility

Choice B is incorrect. This is incorrect because the flowpath is isolated and all pumps are tripped. This is plausible because other systems, like SG blowdown, divert flow path on high rads.

Choice C is incorrect. This is incorrect because the flowpath is isolated and all pumps are tripped. All the sump pumps will trip making this plausible; the addition of tripping the tendon tunnel sump is also logical as this is an input to the F&O sump.

Choice D is correct. The interlock on high rad of OPR05J is to trip all the F & O sump pumps and close ODO030 valve, which is the flowpath isolation to the lime sludge lagoon.

Task No: R-AR-002

Learn Objective: S.AR1-04-B-9

Reference: Big Note AR-1 rev. 10

ILT lesson plan I1-AR-XL-01 rev. 5b

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

24

ID: RG30005-N01

Points: 1.00

Both units are at full power.

- The 0A VC train is in normal operation per BwOP VC-1, STARTUP OF CONTROL ROOM HVAC SYSTEM.
- 0VC032Y, MCR Outside Air, dampers are open.
- The 0B VC train is in standby.

An event that has the potential for an accidental radioactive release in the Unit 2 Turbine Building trackway is reported to the MCR.

- The US directs an RO to monitor control room intake air for elevated radiation trends.
- The RO notes all MCR rad monitor icons on the RMS GRID 2, PROCESS AIR MONITORS, are currently GREEN.

To monitor control room intake air on RMS, the RO will trend the...

- A. 0PR31J or 0PR32J, OUT AIR IN 0A.
- B. 0PR33J or 0PR34J, OUT AIR IN 0B.
- C. 0PR35J or 0PR36J, TURB AIR IN 0A.
- D. 0PR37J or 0PR38J, TURB AIR IN 0B.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 24

Reference Provided: No

History: Bank from Bwd 2016 NRC Exam

RO

Cog level: Memory

K/A- 073G2.4.47 **Importance** RO 4.2 SRO 4.2

073 Process Radiation Monitoring G2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Tier: 2

Group: 1

K/A match: The question requires the examinee to determine the appropriate control room reference instrumentation based on the given plant configuration that must be monitored to recognize trends that could require further action to protect the MCR environment.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is correct. MCR rad monitor icons are green even when their sampled plenums are not online because the sample pumps will continuously sample plenums that have stagnant air flow. With the 0A VC system in normal alignment (outside air intake), the only rad monitors that would have MCR intake air flow through their respective intake plenum is the 0PR31J and 32J.

Choice B is incorrect. This is incorrect because the 0B plenum will not have any flow from outside air as it is in standby. This is plausible because 0PR33J and 34J sample the outside air intake from Unit 2 (0B train). Although the radioactive leak was in the Unit 2 turbine bldg no flow will be seen in that plenum.

Choice C is incorrect. Although the radioactive leak was in the turbine bldg, this plenum would not experience intake air flow unless the 0A VC system was manually or automatically swapped to emergency mode. This plausible because 0PR35J and 36J sample the turbine bldg intake from Unit 1 (0A train) which is in service and the accident happened in the turbine building.

Choice D is incorrect. This is incorrect because the 0B plenum will not have any flow from turbine bldg air as it is in standby and not aligned to emergency intake. This is plausible because the radioactive leak is in the turbine building on the unit 2 side of the turbine building.

Task No: R-AR-002

Learn Objective: 4C.AR-02

Reference: Bwd ILT LP I1-VC-XL-01 rev 3a page 18

Bwd ILT LP I1-AR-XL-01 rev 3c page 29

Big note VC-1 rev. 11

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

25

ID: RS20076-N03

Points: 1.00

A tornado damaged the Braidwood Switchyard causing a loss of all AC power to both Units' SAT's.

- The 2B DG is the ONLY emergency diesel that started and automatically loaded.
- 1BwCA-0.0, LOSS OF ALL AC POWER, is in progress and Bus 142 is crosstied to Bus 242.
- Currently the crew is performing Attachment C, ALTERNATE SX COOLING.

The crew will prepare to crosstie SX by CLOSING BOTH...

- A. 1SX016A AND 2SX016A LOCALLY.
- B. 1SX016A AND 2SX016A MANUALLY from the MCR.
- C. 1SX016B AND 2SX016B LOCALLY.
- D. 1SX016B AND 2SX016B MANUALLY from the MCR.

Answer: A

Answer Explanation
<p>Question: 25 Reference Provided: No History: New for 2018 NRC exam RO Cog level: High K/A-076 K2.08 Importance RO 3.1 SRO 3.3 076 Service Water K2.08 Knowledge of bus power supplies to the following: ESF-actuated MOVs Tier: 2 Group: 1 K/A match: The question requires the examinee to know the power supplies to the service water valves associated with ESF MOVs. 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.</p> <p>Answer explanation and plausibility: Choice A is correct. Per 1BwCA-0.0 attachment C step 1 operators are dispatched to close locally 1 and 2SX016A the deenergized train valves. These valves are powered from their respective ESF bus which is deenergized per the stem, requiring local operation. Choice B is incorrect. This incorrect because both valves will not have power available to manually close them from the control room. This is plausible since bus 242 has power available, and both valves are on one train and there are valves, like 1CV8111, which are powered from the opposite train to provide fail safe isolation. Choice C is incorrect. This is incorrect because both valves have power available to them. This is plausible since bus 242 has power available, and both valves are on one train and there are valves, like 1CV8111, which are powered from the opposite train to provide fail safe isolation. Choice D is incorrect. This is incorrect because both bus 142 and 242 are powered and the RCFs powered from bus 242 will be restored. This will require 2SX016B to remain open for the subsequent steps. This is plausible because both valves have power and are available to be controlled manually from the control room.</p> <p>Task No: R-CA-009 Learn Objective: T.CA1-07 Reference: 1BwCA-0.0 rev 301 Attachment C page 120</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

26

Points: 1.00

Unit 1 is at 100% power.

- During 1B DG surveillance testing, the 1SX169B, DG 1B SX VALVE, was determined to be stuck CLOSED.
- Attempts to open 1SX169B have been unsuccessful, and Tech Spec 3.8.1 has been entered for the 1B DG.

Which of the following Tech Spec Actions must be completed within 1 hour?

- A. Perform 1BwOSR 3.8.1.2-1, 1A DIESEL GENERATOR OPERABILITY SURVEILLANCE.
- B. Verify ONLY the 2B DG is OPERABLE.
- C. Perform 1BwOSR 3.8.1.1, OFFSITE AC POWER AVAILABILITY SURVEILLANCE.
- D. Declare required features supported by the 1B DG INOPERABLE.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 26

Reference Provided: No

History: New for 2018 NRC exam RO Cog level: Memory

K/A-076 G2.2.39 Importance RO 3.9 SRO 4.5

076 Service Water G2.2.39 knowledge of less than or equal to one hour Tech Spec action statements for systems.

Tier: 2

Group: 1

K/A match: The question requires the examinee to know the 1 hour or less Tech Spec's associated with the malfunction of 1SX169B (Service Water).

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect because Tech Spec 3.8.1 required action B.4 requires that the opposite train DGs be verified operable within 24 hours. This is plausible because each unit has 2 DG supplies, one for each bus. Verifying power to one of the two ESF busses guarantees that the minimum required ESF equipment will be available to mitigate accident conditions.

Choice B is incorrect. This is incorrect because Tech Spec 3.8.1 required action B.1 requires that both opposite unit DGs be verified operable. This is plausible because having the opposite unit's same train DG will ensure a backup supply is available to bus 142 through the crosstie from bus 242.

Choice C is correct. This is correct because Tech Spec 3.8.1 required action B.2 requires verifying the requirements of the offsite power availability surveillance within 1 hour.

Choice D is incorrect. This is incorrect because Tech Spec 3.8.1 required action B.2 requires verifying the requirements of the offsite AC power availability surveillance within 1 hour. This is plausible because this is a 4 hour required action B.3 to declare required features supported by the 1B DG inoperable.

Task No: R-DG-015

Learn Objective: 3C.SX-01-A

Reference: Tech Spec 3.8.1

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

27

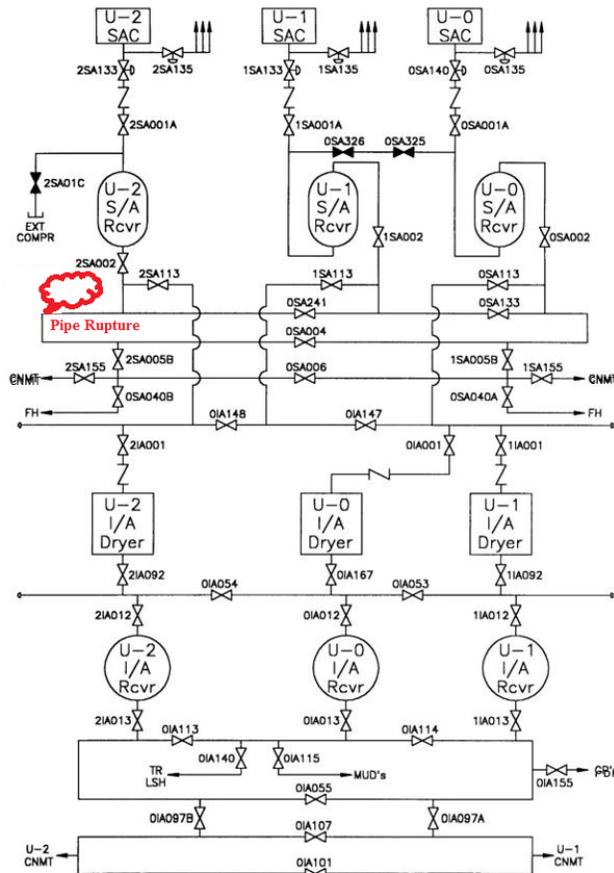
ID: RS10078-001

Points: 1.00

BOTH units are at 100% power.

- Unit 2 Station Air Compressor (SAC) is RUNNING.
- Unit 1 and 0 SACs are in STANDBY.

The station air header pipe ruptured as indicated below.



If NO operator actions are taken, which of the following describes the plant response?

- A. BOTH units will trip when the Instrument Air (IA) header depressurizes due to IA system usage.
- B. BOTH units will trip when the Instrument Air (IA) system depressurizing through the rupture.
- C. ONLY unit 2 will trip when the Instrument Air (IA) system depressurizing through the rupture.
- D. ONLY unit 2 will trip when the Instrument Air (IA) header depressurizes due to IA system usage.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 27

Reference Provided: No

History: Bank question from 2006 NRC exam

RO

Cog level: High

K/A-078K3.03 Importance RO 3.0 SRO 3.4

078 Instrument Air System (IAS) K3.03 Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Cross-tied units.

Tier: 2

Group: 1

K/A match: The question requires the examinee to know that the physical connections (including cross-tied units) and the cause effect relationship between IA and SAS.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is correct. The Station Air ring header in the turbine bldg supplies both units and flows in either direction. (i.e can be supplied by U-1, U-2 or U-0 SAC). Both units would be affected. There are several check valves between the Station Air and Instrument Air systems that would prevent back flow from IA to SA. Therefore, the units would not trip until the IA system air pressure bled down due to system usage and the FRVs went closed.

Choice B is incorrect. This is incorrect because the IA system has accumulators and check valves between the SA and IA system that will provide a delay before the IA loads bleed the IA pressure off. This is plausible since the SA header will blow down rapidly from the main header rupturing. Without knowledge of the IA interconnection having a check valve, this would be likely.

Choice C is incorrect. This is incorrect because the IA system is crosstied between both units and has accumulators and check valves between the SA and IA system. This will provide a delay for the IA loads to bleed the IA pressure off from both units. This is plausible because each unit has its own air system, including compressors, air receivers and dryers. Without knowledge of the system lineup and interconnections between the IA and SA systems this is would be considered likely.

Choice D is incorrect. This is incorrect because the IA system is crosstied between both units and has accumulators and check valves between the SA and IA system. This will provide a delay for the IA loads to bleed the IA pressure off from both units. This is plausible because each unit has its own air system, including compressors, air receivers and dryers. Without knowledge of the system lineup and interconnections between the IA and SA systems this is would be considered likely.

Task No: R-OA-009

Learn Objective: 3D.OA-25-A

Reference: M-55 sheet 1 rev AF; M-54 Sheet 1A rev AD, 0BwOA SEC-4, LOSS OF INSTRUMENT AIR UNIT 0 rev 104

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

28

ID: RS10053-N03

Points: 1.00

Fuel Handlers are defueling the Unit 1 reactor.

An earthquake has occurred.

0BwOA ENV-4 AND _BwOA ENV-4, EARTHQUAKE, have been entered.

- Fuel Handlers report the fuel assembly in containment was dropped from the refueling mast.
- An EO reports finding a crack in the containment wall near pen 66 (1A RH).
- No other issues have been noted after all walk downs were completed.

Which other procedure entry conditions are met?

(1) 1BwOA PRI-11, LOSS OF CONTAINMENT INTEGRITY.

(2) 1BwOA REFUEL-1, FUEL HANDLING EMERGENCY.

(3) 1BwOA REFUEL-2, REFUELING CAVITY OR SPENT FUEL POOL LEVEL LOSS.

- A. ONLY (1)
- B. ONLY (3)
- C. (1) and (2)
- D. (2) and (3)

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 28

Reference Provided: No

History: New question for 2018 NRC exam

RO

Cog level: Memory

K/A-103 K3.03 Importance RO 3.7 SRO 4.1

103 Containment System K3.03 Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under refueling operations.

Tier: 2

Group: 1

K/A match: The question requires the examinee to understand the effect that an event causing a loss of containment integrity will have while in refueling operations by understanding the entry criteria and overall mitigative strategy of the BwOA series of procedures.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because the crack in containment meets entry criteria for 1BwOA PRI-11. This is incorrect because the dropped fuel assembly meets entry criteria for 1BwOA REFUEL-1.

Choice B is incorrect. This is plausible because the crack in containment is at the RH system containment penetration which if the system piping were broken as well would start lowering the refuel cavity level. This is incorrect because the crack in containment meets entry criteria for 1BwOA PRI-11, and the dropped fuel assembly meets entry criteria for 1BwOA REFUEL-1.

Choice C is correct. The dropped fuel assembly meets entry criteria for 1BwOA REFUEL-1, and the crack in containment meets entry criteria for 1BwOA PRI-11.

Choice D is incorrect. This is plausible because the dropped fuel assembly meets entry criteria for 1BwOA REFUEL-1. This is incorrect, none of the entry criteria for 1BwOA REFUEL-2 are met, no radiation conditions are reported and levels are normal based on the report of no other issues noted.

Task No: R-OA-011

Learn Objective: 3D.OA-28-B

Reference: 0BwOA ENV-4 rev. 114; 1BwOA PRI-11 rev. 55; 1BwOA REFUEL-1 rev. 104

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

29

ID: RS20001-N04

Points: 1.00

Which of the following are the power supplies to the associated reactor trip BYPASS breakers tripping mechanisms?

- | | | | |
|----|-----|-----------------|--------------------|
| A. | | <u>UV Coils</u> | <u>Shunt Coils</u> |
| | BYA | Trn A SSPS | DC Bus 111 |
| | BYB | Trn B SSPS | DC Bus 112 |
| B. | | <u>UV Coils</u> | <u>Shunt Coils</u> |
| | BYA | Trn B SSPS | DC Bus 112 |
| | BYB | Trn A SSPS | DC Bus 111 |
| C. | | <u>UV Coils</u> | <u>Shunt Coils</u> |
| | BYA | Trn B SSPS | DC Bus 111 |
| | BYB | Trn A SSPS | DC Bus 112 |
| D. | | <u>UV Coils</u> | <u>Shunt Coils</u> |
| | BYA | Trn A SSPS | DC Bus 112 |
| | BYB | Trn B SSPS | DC Bus 111 |

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 29

Reference Provided: No

History: New question for 2018 NRC exam

RO

Cog level: memory

K/A-001K2.02 Importance RO 3.6 SRO 3.7

001 Control Rod Drive System K2.02 Knowledge of bus power supplies to the following: one line diagram of power supply to trip breakers.

Tier: 2

Group: 2

K/A match: The question requires the examinee to know the power supplies to the trip coils for the reactor trip and bypass breakers.

10CFR55.41(b)(6) Design, components, and functions of reactivity control mechanisms and instrumentation.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect because the reactor trip bypass breakers UV Coil is powered from the opposite train of SSPS to provide for the ability to test a train of SSPS while the reactor is still critical. This is plausible because the crossing of train power supplies and different coils responding to different trip signals is very often confused.

Choice B is incorrect. This is incorrect because the reactor trip bypass breakers UV Coil is powered from the opposite train of SSPS to provide for the ability to test a train of SSPS while the reactor is still critical. This is plausible because the crossing of train power supplies and different coils responding to different trip signals is very often confused.

Choice C is correct. The reactor trip bypass breakers UV Coil is power from the opposite train of SSPS to provide for the ability to test a train of SSPS while the reactor is still critical.

Choice D is incorrect. This is incorrect because the reactor trip bypass breakers UV Coil is powered from the opposite train of SSPS to provide for the ability to test a train of SSPS while the reactor is still critical. This is plausible because the crossing of train power supplies and different coils responding to different trip signals is very often confused.

Task No: R-RP-007

Learn Objective: 3C.EF-02-A

Reference: 20E-1-4030RD06 rev S, 4030RD07 rev U

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

30

ID: RS20011-N04

Points: 1.00

Unit 1 is at 98% power.

- Tave - Tref deviation is 0°F.
- 1FK-110, Boric Acid Flow Controller, set point is set HIGHER than the flowrate needed to maintain current RCS boron concentration.

Subsequently:

- A Unit 1 VCT auto makeup occurs.
- Control Bank D rods withdraw in AUTO from 215 to 218 steps following the automatic makeup.

No operator actions have been taken.

After the plant stabilizes following the RMCS auto makeup and rod withdrawal, Tave will be (1) and PZR level will be (2).

- A. (1) LOWER
(2) LOWER
- B. (1) LOWER
(2) HIGHER
- C. (1) HIGHER
(2) LOWER
- D. (1) HIGHER
(2) HIGHER

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 30

Reference Provided: No

History: Modified from BWD ILT Bank question RO Cog level: High

K/A-011 A3.01 Importance RO 2.8 SRO 2.8

011 Pressurizer Level Control System A3.01 Ability to monitor automatic operation of the PZR LCS, including: Boration/dilution.

Tier: 2

Group: 2

K/A match: The question requires the examinee to determine the integrated plant response to the boration and based on that response the overall response of the PZR LCS and its control setpoint.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Answer explanation and plausibility:

Choice A is correct. The excess boron will cause temperature to lower until the deviation is greater than 1.5°F, and rods begin to step out. Rods will step out until temperature deviation is less than 1°F. This new lower stable temperature will correspond to a new lower PZR LCS setpoint level.

Choice B is incorrect, the lower Tavg will result in a lower PZR LCS setpoint level. The lower avg temperature is correct combined with the temporary level rise from the RCS temperature rise caused by the rod withdrawal, make this a plausible distractor.

Choice C is incorrect, the lockup on the rod control system will cause the rods to move out at a -1.5°F Terr and stop moving at a -1°F Terr. The lower than original PZR level is correct with the idea that rods stepping out raises temperature make this a plausible distractor.

Choice D is incorrect, the lockup on the rod control system will cause the rods to move out at a -1.5°F Terr and stop moving at a -1°F Terr. The temporary level rise from the RCS temperature rise caused by the rod withdrawal, with the idea that rods stepping out raises temperature make this a plausible distractor.

Task No: R-RY-015

Learn Objective: 3S.RY-01

Reference: I1-RY-XL-01

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

31

ID: RS20015-N04

Points: 1.00

Unit 1 is in Mode 6.

- Fuel moves are in progress.
- BOTH Westinghouse Source Range Monitors N-31 AND N-32 are DE-ENERGIZED.

Fuel moves...

- A. CANNOT continue until at least ONE Westinghouse Source Range Detector is restored.
- B. CANNOT continue until BOTH Westinghouse Source Range Detectors are restored.
- C. CAN continue with ONLY ONE Gamma Metric Post Accident Neutron Flux Monitor OPERABLE.
- D. CAN continue ONLY if BOTH Gamma Metric Post Accident Neutron Flux Monitors are OPERABLE.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 31

Reference Provided: N

History: New question for 2018 NRC exam

RO

Cog level: Memory

K/A- 015K3.03 **Importance** RO 2.7 SRO 3.4

015 Nuclear Instrumentation K3.03 Knowledge of the effect that a loss or malfunction of the NIS will have on the following: Fuel handling system

Tier: 2

Group: 2

K/A match: The question requires the examinee to have knowledge of the effect a loss of NIS will have on 1 hour or less Tech Spec 3.9.3 and what constitutes a source range monitor.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect because Tech Spec 3.9.3 allows for the use of post-accident neutron flux monitors instead of the source range detectors. This is plausible since Tech Spec 3.9.3 has a condition for 1 required source range neutron flux monitor being inoperable.

Choice B is incorrect. This is incorrect because Tech Spec 3.9.3 allows for the use of post-accident neutron flux monitors instead of the source range detectors. This is plausible since Tech Spec 3.9.3 has a condition for 2 required source range neutron flux monitor being inoperable.

Choice C is incorrect. This is incorrect because Tech Spec 3.9.3 allows for the use of post-accident neutron flux monitors instead of the source range detectors, but 2 monitors are required. This is plausible since Tech Spec 3.9.3 has a condition for 1 required source range neutron flux monitor being inoperable.

Choice D is correct. Tech Spec 3.9.3 requires 2 source range detectors. This is defined as any two the installed westinghouse source range detectors or the post-accident neutron flux monitors. The candidate most know that 2 detectors can be any combination of 2 out of the 4 detectors from the bases definition of source range detectors.

Task No: R-OA-053

Learn Objective: T.AM12-05

Reference: Tech Spec 3.9.3, Bases 3.9.3

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

32

ID: RS20002-N04

Points: 1.00

Unit 1 is at 100% power.

- At 1300, pressurizer level is 59.6%.
- At 1302, pressurizer level is 59.0%.
- RCS temperature is stable.
- Containment pressure is stable.

The crew will ___(1)___ in accordance with ___(2)___.

- A. (1) remove the 45 GPM letdown orifice from service
(2) 1BwOA PRI-1, EXCESSIVE PRIMARY PLANT LEAKAGE UNIT 1
- B. (1) remove the 45 GPM letdown orifice from service
(2) 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION UNIT 1
- C. (1) trip the reactor and actuate SI
(2) 1BwOA PRI-1, EXCESSIVE PRIMARY PLANT LEAKAGE UNIT 1
- D. (1) trip the reactor and actuate SI
(2) 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION UNIT 1

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 32

Reference Provided: No

History: new for 2018 NRC exam

RO

Cog level: High

K/A-002A2.01 Importance RO 4.3 SRO 4.4

0002 Reactor Coolant A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of coolant inventory

Tier: 2

Group: 2

K/A match: This question requires the examinee to predict the affect on the pressurizer level system by determining the size of the leak. This prediction allows the candidate to identify that a reduction in letdown will be sufficient to maintain RCS inventory and based on that information determine that entry criteria for 1BwOA PRI-1 are met and the mitigating strategy will stabilize the plant.

10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Answer explanation and plausibility:

Choice A is correct. The leak rate based on losing .6% in 2 minutes is approximately 40 GPM $((.6\% * 128 \text{ Gallon}/\%)/(2 \text{ min}))$. This is within the capacity of a single charging pump to handle and would require entry to 1BwOA PRI-1. The actions in 1BwOA PRI-1 include securing the 45 GPM letdown orifice. With the reduced letdown flow, charging is more than the leakage and level will be stabilized. By continuing with actions in 1BwOA PRI-1, the leak will be located and isolated if possible.

Choice B is incorrect. This is incorrect because once the 45 GPM letdown orifice is isolated PZR level will be stabilized and steps in 1BwOA PRI-1 will be taken to isolate the leak if possible or perform an orderly shutdown per the GP procedures. This is plausible because removing the 45 GPM letdown orifice is the correct action that would be taken, and if pressurizer level drop can't be stopped, you would trip the reactor and initiate safety injection.

Choice C is incorrect. This is incorrect because the leak rate of approximately 40 GPM is within the capacity of a charging pump and a reactor trip is not required. 1BwOA PRI-1 will be the procedure used to mitigate the leakage, making this plausible. 1BwOA PRI-1 does have actions for tripping the reactor if pressurizer level cannot be maintained.

Choice D is incorrect. This is incorrect because the leak rate of approximately 40 GPM is within the capacity of a charging pump and a reactor trip is not required. Steps in 1BwOA PRI-1 will be taken to isolate the leak if possible or perform an orderly shutdown per the GP procedures. If you incorrectly calculate that the leakage to be greater than the capacity of a charging pump, you would trip the reactor and initiate safety injection, making this plausible.

Task No: R-OA-060

Learn Objective: 3D.OA-37-E

Reference: 1BwOA PRI-1 rev 107

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

33

ID: RS20045-N02

Points: 1.00

Unit 1 is 100% power.

- A reactor trip/turbine trip occurred.
- Some turbine control valves did NOT close (are stuck open).
- Operators attempted to manually trip the turbine from both the 1PM02J and OWS graphic 5512.

Which of the following screen shots from DEHC Graphic Panel 5501 require the operator to take further manual action to trip the turbine?

A.

MS5005C	MS5006C	MS5006A	MS5005A	
TV3 OPEN	GV3 OPEN	GV1 OPEN	TV1 OPEN	LEFT
TV3 CLOSED	GV3 CLOSED	GV1 CLOSED	TV1 CLOSED	
TV4 OPEN	GV4 OPEN	GV2 OPEN	TV2 OPEN	RIGHT
TV4 CLOSED	GV4 CLOSED	GV2 CLOSED	TV2 CLOSED	
MS5005D	MS5006D	MS5006B	MS5005B	

B.

MS5005C	MS5006C	MS5006A	MS5005A	
TV3 OPEN	GV3 OPEN	GV1 OPEN	TV1 OPEN	LEFT
TV3 CLOSED	GV3 CLOSED	GV1 CLOSED	TV1 CLOSED	
TV4 OPEN	GV4 OPEN	GV2 OPEN	TV2 OPEN	RIGHT
TV4 CLOSED	GV4 CLOSED	GV2 CLOSED	TV2 CLOSED	
MS5005D	MS5006D	MS5006B	MS5005B	

C.

MS5005C	MS5006C	MS5006A	MS5005A	
TV3 OPEN	GV3 OPEN	GV1 OPEN	TV1 OPEN	LEFT
TV3 CLOSED	GV3 CLOSED	GV1 CLOSED	TV1 CLOSED	
TV4 OPEN	GV4 OPEN	GV2 OPEN	TV2 OPEN	RIGHT
TV4 CLOSED	GV4 CLOSED	GV2 CLOSED	TV2 CLOSED	
MS5005D	MS5006D	MS5006B	MS5005B	

D.

MS5005C	MS5006C	MS5006A	MS5005A	
TV3 OPEN	GV3 OPEN	GV1 OPEN	TV1 OPEN	LEFT
TV3 CLOSED	GV3 CLOSED	GV1 CLOSED	TV1 CLOSED	
TV4 OPEN	GV4 OPEN	GV2 OPEN	TV2 OPEN	RIGHT
TV4 CLOSED	GV4 CLOSED	GV2 CLOSED	TV2 CLOSED	
MS5005D	MS5006D	MS5006B	MS5005B	

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer: C

Answer Explanation

Question: 33

Reference Provided: No

History: Modified question for 2018 NRC exam from RE10007-N01.Bank (Braidwood NRC Exam 2011) RO **Cog level:** High

K/A-045 G2.4.1 Importance RO 4.6 SRO 4.8

045 Main Turbine Generator G2.4.1 Knowledge of EOP entry conditions and immediate action steps.

Tier: 2

Group: 2

K/A match: The question requires the examinee to know the indications for the main turbine that would require immediate actions in the EOPs.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect because the immediate action step of 1BwEP-0 step 2 requires action if any throttle and governor valve are open providing a possible steam flow path to the main turbine. This is plausible because two valves are open on the main turbine. 1 GV on each side of the HP turbine and could be incorrectly interpreted as a flow path.

Choice B is incorrect. This is incorrect because the immediate action step of 1BwEP-0 step 2 requires action if any throttle and governor valve are open providing a possible steam flow path to the main turbine. This is plausible because two valves are open on the main turbine. 2 TVs on the same side of the HP turbine and could be incorrectly interpreted as a flow path.

Choice C is correct. Step 2, "VERIFY TURBINE TRIP:" of 1BwEP-0 first checks that all throttle and governor valves are closed. If not, the RNO requires a manual trip and "if any throttle AND governor valve are open, then manually isolate steam."

Choice D is incorrect. This is incorrect because the immediate action step of 1BwEP-0 step 2 requires action if any throttle and governor valve are open providing a possible steam flow path to the main turbine. This is plausible because two valves are open on the main turbine, 2 TVs one on each side of the HP turbine and could be incorrectly interpreted as a flow path.

Task No: R-EP-023

Learn Objective: 4D.EP-28

Reference: 1BwEP-0 rev 302

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

34

ID: RS20035-N05

Points: 1.00

Unit 2 tripped from 100% power due to a inadvertent Main Steam Line Isolation actuation.

- Immediately after the trip, a Loss of Offsite Power occurred.
- The transient caused SG pressures to rise past the SG PORV opening setpoint.
- The 2B SG PORV failed closed and did not open.
- Currently, 2B SG pressure is 1200 PSIG.

The 2B SG will have a total of ____ Safety Valves OPEN.

- A. 1
- B. 2
- C. 3
- D. 4

Answer: B

Answer Explanation

Question: 34

Reference Provided: No

History: New for 2018 NRC exam

RO

Cog level: Memory

K/A-035 K6.02 Importance RO 3.1 SRO 3.5

035 Steam Generator K6.02 Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: Secondary PORV

Tier: 2

Group: 2

K/A match: The question requires the examinee to know the effect on the SG safety valves for the failure of the SG PORV.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect because the SG safety set points are 1175,1190,1205,1220,1235. This is plausible because confusing set points is a common error.

Choice B is correct. This is correct because the SG safety set points are 1175,**1190**,1205,1220,1235. Pressure is above the set point of the first and second SG safety.

Choice C is incorrect. This is incorrect because the SG safety set points are 1175,1190,1205,1220,1235. This is plausible because confusing set points is a common error.

Choice D is incorrect. This is incorrect because the SG safety set points are 1175,1190,1205,1220,1235. This is plausible because confusing set points is a common error.

Task No: R-MS-007

Learn Objective: S.MS1-08-D

Reference: M-120 sheet 1 rev AO

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

35

ID: RS20041-N04

Points: 1.00

Unit 1 is performing a plant startup with RCS temperature being maintained by steam dumps in Steam Pressure Mode.

- 1UI-500, STM DUMP DEMAND, is indicating 10% and stable.

1PK-507 will be indicating Main Steam Header Pressure of _____ psig.

- A. 1092
- B. 1102
- C. 1115
- D. 1201

Answer: A

Answer Explanation

Question: 35

Reference Provided: No

History: New for 2018 NRC exam RO Cog level: High

K/A-041A4.05 Importance RO 3.1 SRO 3.3

041 Steam Dump System A4.05 Ability to manually operate and/or monitor in the control room: Main steam header pressure.

Tier: 2

Group: 2

K/A match: The question requires the examinee to have knowledge of the operation of the main steam header pressure control system and the ability to monitor operations based on multiple indications.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is correct. This is correct because the 1PK-507 controller has an integral function that will continue to drive the error signal higher until pressure is restored to its original set point (1092) with a STM DUMP DEMAND greater than 0%.

Choice B is incorrect. This is incorrect because the 1PK-507 controller has an integral function that drives the control signal back to the original setpoint. This is plausible because 1UI-500 indicates 1% demand for each 1 psig above the setpoint (1092), so a 10% error would indicate a pressure 10 psig above the 1092 psig setpoint (1102) without the integral function working.

Choice C is incorrect. This is incorrect because the 1PK-507 controller has an integral function that will continue to drive the error signal higher until pressure is restored to its original set point (1092) with a STM DUMP DEMAND greater than 0%. This is the normal opening setpoint for the SG PORV making this plausible.

Choice D is incorrect. This is incorrect because the 1PK-507 controller has an integral function that will continue to drive the error signal higher until pressure is restored to its original setpoint (1092) with a STM DUMP DEMAND greater than 0%. This is a plausible value because the value is 110% above the no load setpoint.

Task No: R-DU-001

Learn Objective: 4C.DU-01

Reference: Big Note MS-4 rev 15

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

36

ID: BWLI-GW1-048

Points: 1.00

A full Gas Decay Tank (GDT) in STORAGE has an inadvertent lifting of its relief valve.

The GDT contents will be released to the (1) and the release is (2) by the OPR02J, GDT Process Rad Monitor.

- A. (1) Plant Vent Stack
(2) monitored
- B. (1) Plant Vent Stack
(2) NOT monitored
- C. (1) GDT cubicle
(2) monitored
- D. (1) GDT cubicle
(2) NOT monitored

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 36

Reference Provided: No

History: New for 2018 NRC exam

RO

Cog level: Memory

K/A-071A1.06 Importance RO 2.5 SRO 2.8

071 Waste Gas Disposal A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Waste Gas Disposal System operating the controls including: Ventilation system

Tier: 2

Group: 2

K/A match: The question requires the examinee to know where the waste gas disposal system reliefs tie into ventilation and how they are monitored.

10CFR55.41(b)(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Answer explanation and plausibility:

Choice A is correct. The relief valves for the waste gas storage tanks lift in the normal release path which is through the vent stack. The relief valves discharge downstream of the release path isolation valve but upstream of the 0PR02J which will provide monitoring of the release.

Choice B is incorrect. This is incorrect because the release path is monitored by the 0PR02J. This is plausible since the normal release path for the gas decay tanks has an isolation valve that isolates on high radiation and the relief valves input to the header downstream of the isolation valve.

Choice C is incorrect. This is incorrect because the gas decay tanks relief valves are piped directly to the release line downstream of the release isolation valve. This is plausible since the release is monitored and would be monitored from auxiliary building ventilation drawing through the GDT cubicle. Many tanks have reliefs that lift directly to their environment.

Choice D is incorrect. This is incorrect because the gas decay tanks relief valves are piped directly to the release line downstream of the release isolation valve. This is plausible since many tanks have reliefs that lift directly to their environment.

Task No: R-GW-001

Learn Objective: S.GW1-07

Reference: M-69 sheet 1 rev AX

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

37

ID: RS2072-N14-09

Points: 1.00

Unit 1 is at 100% power.

Containment atmosphere purge is in progress.

- The 1VQ04C, Cnmt Mini-Flow Purge Supply Fan, is running.
- The 1VQ05C, Cnmt Mini-Flow Purge Exhaust Fan, is running.
- The 1AR012J, Containment Fuel Handling Incident Monitor, alarms RED on the RMS.

What AUTOMATIC actions, if any, will occur?

- A. NO automatic action occur, manual actions are required to terminate the release.
- B. ONLY 'A' Train mini-flow purge valves close and ONLY 1VQ04C trips.
- C. ONLY 'B' Train mini-flow purge valves close and BOTH 1VQ04C and 1VQ05C trip.
- D. BOTH 'A' and 'B' Train mini-flow purge valves close and BOTH 1VQ04C and 1VQ05C trip.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 37

Reference Provided: No

History: Bank 2014 NRC exam RO **Cog level:** Memory

K/A-072 K4.01 Importance RO 3.3 SRO 3.6

072 Area Radiation Monitoring (ARM) System K4.01 Knowledge of ARM system design feature(s) and/or interlock(s) which provide for the following: containment ventilation isolation.

Tier: 2

Group: 2

K/A match: Examinee must have knowledge of the containment ventilation interlocks associated with high radiation levels of the ARM System.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect, the 1VQ05C, Cnmt Mini Flow Purge Exhaust Fan, does not trip. This is plausible because the 1PR01J, containment purge effluent radiation monitor, has no automatic action.

Choice B is incorrect. This is incorrect, the A Train mini-flow purge valves go closed when the 1AR011J alarms (turns red). This is plausible because the examinee may conclude the 1AR012J closes the A Train mini-flow purge valves as well, similar to SI or other actuations that have isolation from each train of actuation signal to each valve.

Choice C is correct. When the HIGH alarm comes in. The 1AR012J on the RMS will turn RED for that monitor. When the 1AR012J exceeds the HIGH alarm, only the B Train mini-flow purge valves go closed (1VQ003, 1VQ004B, and 1VQ005B will go closed, 1VQ001B and 1VQ002B will also get a closed signal, but the valves are abandoned in place). 1VQ04C, Cnmt Mini-Flow Purge Supply, and 1VQ05C, Cnmt Mini-Flow Purge Exhaust Fan, will trip when the B train mini-flow purge valves go closed. 1VQ004B going closed trips the 1VQ04C, Cnmt Mini-Flow Purge Supply Fan, and when 1VQ005B goes closed the 1VQ05C, Cnmt Mini-Flow Purge Exhaust Fan, will trip.

Choice D is incorrect. This is incorrect because the 1AR012J only closes the B train valves. This is plausible because the examinee may conclude the 1AR012J closes both the A and B Train mini-flow purge valves similar to SI or other actuations that have isolation from each train of actuation signal to each valve.

Task No: R-VQ-006

Learn Objective: S.AR1-04-A-1

Reference: BwAR 4-1AR012J rev 7 and BwAR 4-1AR011J rev 7 and Big Note VP-2

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

38

ID: RS20076-N01

Points: 1.00

Both units are at 100% power.

- 1A Circ Water Pump forebay is drained for silt removal.
- 0SX115A, UPST ISOL TO SX PUMPS TRAIN A SUCTION HDR FROM 1A BAY, is danger tagged CLOSED.
- All other SX system equipment is in normal alignment.

An event occurs that causes debris in the 1B Circ Water Pump forebay to block flow at the inlet to the SX suction piping.

ONLY the 1B Circ Water Pump forebay is affected.

ALL SX pumps are running.

The NSO will see cavitation from the _____ SX Pump(s).

- A. 1A (ONLY)
- B. 1A AND 2A
- C. 1A AND 1B
- D. 1B AND 2B

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 38

Reference Provided: No

History: Bank question 2009 NRC exam

RO

Cog level: High

K/A-075 K1.08 Importance RO 3.2 SRO 3.2

075 Circulating Water K1.08 Knowledge of the physical connections and/or cause effect relationships between the circulating water system and the following system: essential service water.

Tier: 2

Group: 2

K/A match: This question requires the examinee to understand the connection between the circulating water pump forebays and the suction to the SX pump headers.

10CFR55.41(b)(4) Secondary coolant and auxiliary systems that affect the facility.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect because the unit 1 header supplies the A train suction for both units. This is plausible since the only affected units circulating water system is on unit 1 and without knowledge of the connections between the circulation water system and SX, this could incorrectly be seen as to only affect unit 1 SX A train.

Choice B is correct. The SX suction header is designed such that the 3 U-1 forebays each combine to a common suction header to the 1A and 2A SX pumps. Likewise the U-2 forebays combine for the "B" train suction header. The system is designed such that it can operate full capacity with any 1 of the 3 forebays out of service and the respective 0SX115 valve closed. If debris were to block a second forebay on the unit 1, it would affect the train A SX pumps on both units.

Choice C is incorrect. This is incorrect because the unit 1 header supplies the A train suction for both units. This is plausible since the only affected units circulating water system is on unit 1 and without knowledge of the connections between the circulation water system and SX, this could incorrectly be seen as to only affect unit 1 SX pumps.

Choice D is incorrect. This is incorrect because the unit 1 header supplies the A train suction for both units. This is plausible since the affected circulating water forebay is 1B. The examinee could incorrectly assume that the 1B CW forebay provides the B suction header since the 1A supplies the A header.

Task No: R-SX-006

Learn Objective: 4C.SX-03

Reference: M-42 sheet 1A rev BL; sheet 1B rev BD

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

39

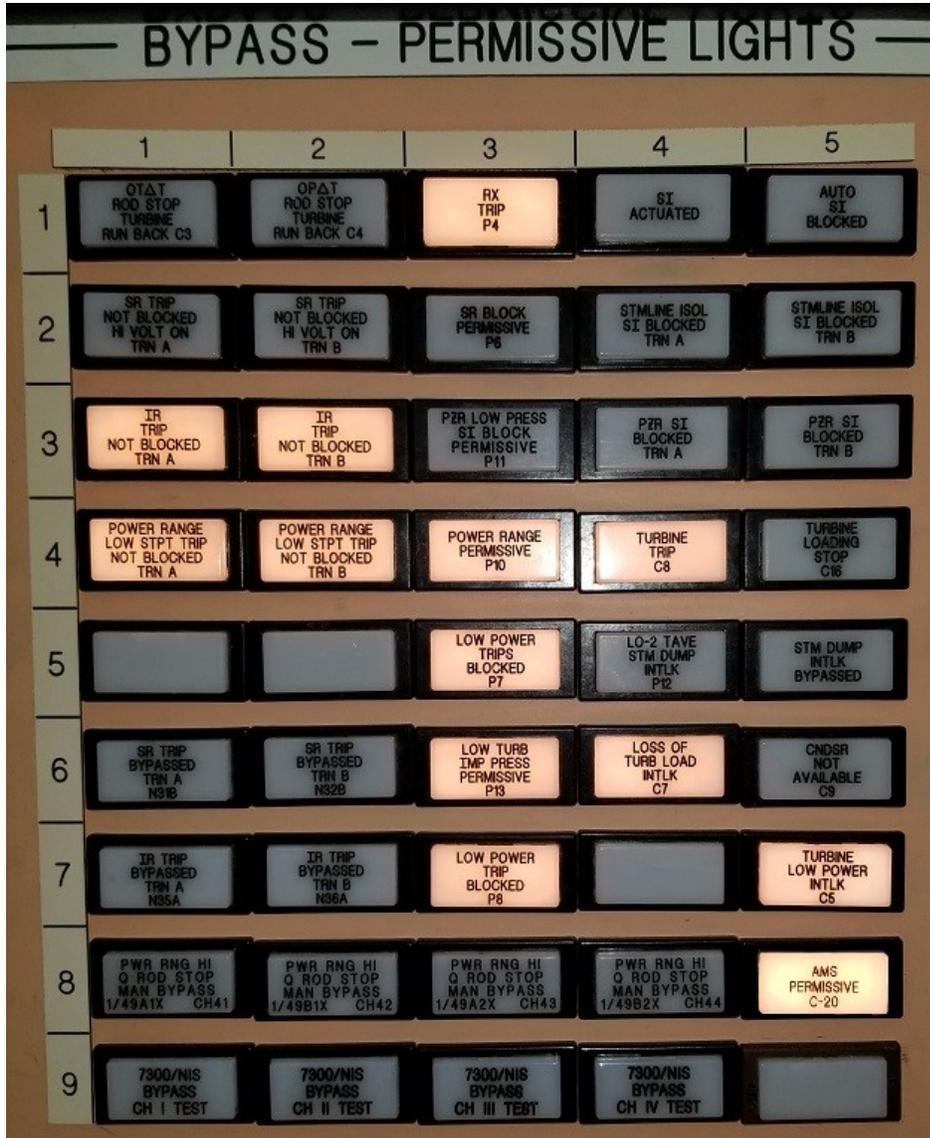
ID: RE10007-N04

Points: 1.00

Unit 1 tripped from 100% power 20 minutes ago.

- All alarms have been reset and acknowledged.
- No alarms are fast flashing.

Which of the following Bypass- Permissive lights are NOT indicating correctly for the current plant conditions?



- A. BOTH trains of "SR TRIP NOT BLOCKED HI VOLT ON" should be lit.
- B. "LOSS OF TURB LOAD INTLK C7" should NOT be lit.
- C. BOTH trains of "SR TRIP BYPASSED" should be lit.
- D. "AMS PERMISSIVE C-20" should NOT be lit.

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer: A

Answer Explanation

Question: 39

Reference Provided: No

History: New for 2018 NRC exam RO Cog level: High

K/A- EPE 007 EK2.03 **Importance** RO 3.5 SRO 3.6

EPE 007 Reactor Trip EK2.03 Knowledge of the interrelationship between a reactor trip and the following: Reactor trip status panel.

Tier: 1

Group: 1

K/A match: The question requires the examinee to understand the relationship between post reactor trip indication and the status of the BYPASS PERMISSIVE (reactor trip status) panel.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is correct. 20 minutes post trip, the source range detectors should be energized which will cause the "SR TRIP NOT BLOCKED HI VOLT ON" status lights to be lit.

Choice B is incorrect, C-7 has a latching relay, once it is engaged it will require manual action to reset the loss of load interlock. This control requires a load drop over a 2 minute time frame and can be manually reset once it has been greater than 2 minutes since the turbine tripped, making this a plausible distractor.

Choice C is incorrect, the SR detectors do not have their trips bypassed unless they are being tested or have a failure, which is not the case in this question. The SR detectors should be energized at 20 minutes post trip and the trips are normally blocked when the SR detectors are not energized. In both the bypass and blocked state, the SR detector is not providing reactor protection, making this a plausible distractor.

Choice D is incorrect, when turbine power is below 30%, C-20 will be lit. The C-20 AMS control has a 6 minute time delay to prevent AMS from disarming it too soon on a trip. This and the dark board concept causing C-20 to be NOT lit when it is armed, making this a plausible distractor.

Task No: R-EP-011

Learn Objective: 4C.NI-04

Reference: I1-MI-XL-11 Accident response of instrumentation rev 3a

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

40

ID: RE10008-N02

Points: 1.00

Unit 1 Reactor trip and SI occurred.

- 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, step 2, CHECK IF SGs SECONDARY PRESSURE BOUNDARIES ARE INTACT, is in progress.
- All SG pressures are 1050 psig and stable.
- PZR level has risen continuously.
- RCS pressure has lowered continuously.

The failure is a leak through the ...

- A. open RX vessel head vent valve.
- B. PZR liquid space sample system.
- C. open PZR safety valve.
- D. 1B RCS hot leg isolation valve body.

Answer: C

Answer Explanation

Question: 40

Reference Provided: No

History: 2014 NRC exam RO **Cog level:** High

K/A- E/APE 008AK2.01 **Importance** RO 2.7 SRO 2.7

E/APE 008 Pressurizer Vapor Space Accident

AK2.01 Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Valves

Tier: 1

Group: 1

K/A match: Meets K/A, requires examinee to evaluate the indications and determine that the only answer that meets this is related to a of a vapor space accident through the PZR safety valve.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect, PZR level will also drop. This is plausible because a leak on the reactor vessel head is a LOCA resulting in lowering PZR pressure.

Choice B is incorrect, PZR level will also drop. A leak on the PZR liquid space sample system is a LOCA resulting in lowering PZR pressure, making this plausible.

Choice C is correct. PZR pressure lowering and level rising is indicative of a vapor space LOCA. PZR vapor space connections are PZR safeties, PORVs, Sample lines and Instrumentation.

Choice D is incorrect, PZR level will also drop. A leak on the 1B RCS hot leg isolation valve body is a LOCA resulting in lowering PZR pressure, making this plausible.

Task No: R-EP-032

Learn Objective: T.EP02-09.

Reference: I1-EP-XL-02 rev. 16a.

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

41

ID: RE10009-N05

Points: 1.00

In regards to core cooling, the worst case location for a small break LOCA is a break in an RCS (1) leg,

because a loop seal (2) formed.

- A. (1) cold
(2) is
- B. (1) cold
(2) is NOT
- C. (1) hot
(2) is
- D. (1) hot
(2) is NOT

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 41

Reference Provided: No

History: New for 2018 NRC exam RO **Cog level:** Memory

K/A-EPE09 EK1.01 Importance RO 4.2 SRO 4.7

EPE09 Small Break LOCA EK1.01 Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Natural circulation and cooling, including reflux boiling.

Tier: 1

Group: 1

K/A match: The question requires the examinee to understand the operational implications of reflux boiling on a small break LOCA.

10CFR55.41(b)(8) Components, capacity, and functions of emergency systems.

Answer explanation and plausibility:

Choice A is correct. The cold leg break will allow the formation of a loop seal between the SG and the RCP. This loop seal is maintained full by the reflux boiling occurring in the SG U tubes. This impedes the direct venting of steam from the reactor vessel and lengthens the time to lower pressure and raise ECCS injection flow. For the loop seal to clear, level will have to lower below the top of the fuel before there is a large enough pressure difference to force the water in the loop seal out and clear the release path for steam and energy to lower RCS pressure. This allows the ECCS flow to rise and meet or exceed break flow and restore adequate core cooling.

Choice B is incorrect, the loop seal will form due to reflux boiling and allow uncovering of the reactor core. This is plausible since the cold leg is the worst location in regards to core cooling for a break.

Choice C is incorrect, the cold leg break will allow the formation of a loop seal between the SG and the RCP. This loop seal is maintained full by the reflux boiling occurring in the SG U tubes. This impedes the direct venting of steam from the reactor vessel and lengthens the time to lower pressure and raise ECCS injection flow. A loop seal between the SG and the RCP is formed in a hot leg break. The small break LOCA in the hot leg is the worst-case accident for an INTEGRITY event, making this a plausible distractor.

Choice D is incorrect, a cold leg break forms a loop seal between the SG and the RCP delaying the release of energy to lower pressure and raise the ECCS flow. The small break LOCA in the hot leg break is the worst-case accident for an INTEGRITY event. The hot leg small break provides ample flow to keep the core cooled, making this a plausible distractor.

Task No: R-FR-008

Learn Objective: T.MI07-02

Reference: I1-MI-XL-07, MCD 2.2 rev 2c

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

42

ID: RE10011-N04

Points: 1.00

Unit 1 is at 100% power.

- Excess Letdown is in service due to normal letdown being unavailable.
- A Unit 1 containment release is in progress.
- All other systems are normally aligned.

An RCS LOCA is diagnosed.

- The reactor is tripped and SI actuated.
- The crew is performing 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION.

During steps 8 and 9, checking Group 3 CNMT isol monitor lights and Group 6 CNMT VENT isol monitor lights, the following lights are NOT lit.

- 1CC9437B, CC FROM EXC LTDWN HX ISOL VLV
- 1CV8100, SEAL WTR RETURN CNMT ISOL VLV
- 1VQ005C, MINI-FLOW PRG EXH OUTSIDE ISOL VLV
- 1IA065, INST AIR OUTSIDE ISOL VLV

All OTHER Group 3 CNMT isol and Group 6 CNMT VENT isol monitor lights are LIT.

Which of the following systems has a containment penetration that is NOT isolated?

- A. Component Cooling Water (CC)
- B. Seal Water Return (CV)
- C. Containment Mini-Purge (VQ)
- D. Instrument Air (IA)

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 42

Reference Provided: No

History: Bank 2009 NRC exam RO **Cog level:** High

K/A-EPE011EA1.05 Importance RO 4.4 SRO 4.4

EPE011 Large Break LOCA EA1.07 Ability to operate and monitor the following as they apply to a Large Break LOCA: Containment isolation system.

Tier: 1

Group: 1

K/A match: The question requires examinee ability to monitor isolation valves as they apply to loss of containment integrity (determine which path is not isolated).

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is correct. CC to/from Excess Letdown HX has single valve isolation on the inlet and outlet flowpath.

Choice B is incorrect. This is incorrect because the CV (seal return) outlet path also has 1CV8112 (& 1CV8113 check valve) inside containment. This is plausible because this is the outside containment isolation valve and 1BwEP-0 step 8 RNO directs that "if any valve(s) located outside cnmt can NOT be closed manually, then close it locally", which could lead the examinee to incorrectly believe that the isolation valve must be closed to isolate the flow path.

Choice C is incorrect. This is incorrect because the VQ (mini flow) outlet path also has 1VQ005A valve inside containment. This is plausible because this is an outside containment isolation valve and 1BwEP-0 step 9 RNO directs that "Stop any running VQ fans. Manually close VQ isol valve(s) as necessary", which could lead the examinee to incorrectly believe that the isolation valve must be closed to isolate the flow path.

Choice D is incorrect. This is incorrect because the IA path also has 1IA066 inside containment. This is plausible because this is the outside containment isolation valve and 1BwEP-0 step 8 RNO directs that "if any valve(s) located outside cnmt can NOT be closed manually, then close it locally", which could lead the examinee to incorrectly believe that the isolation valve must be closed to isolate the flow path.

Task No: R-OA-011

Learn Objective: 3S.PC-01

Reference: Bwd Big Note CV-1 rev. 16

Bwd Big Note CC-2 rev. 7

Bwd Big Note VP-2 rev. 9

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

43

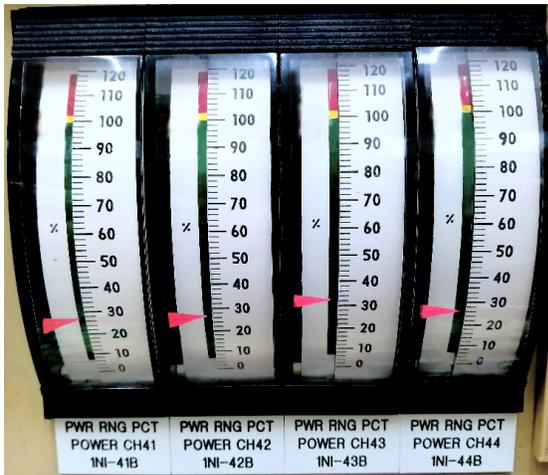
ID: RE10015-N05

Points: 1.00

The LOOP 1A RCS FLOW indicators are SLOWLY LOWERING due to a degrading RCP impeller.

Given the following PRNI indications, which case(s) would result in an automatic reactor trip at the Reactor Coolant Flow - Low setpoint?

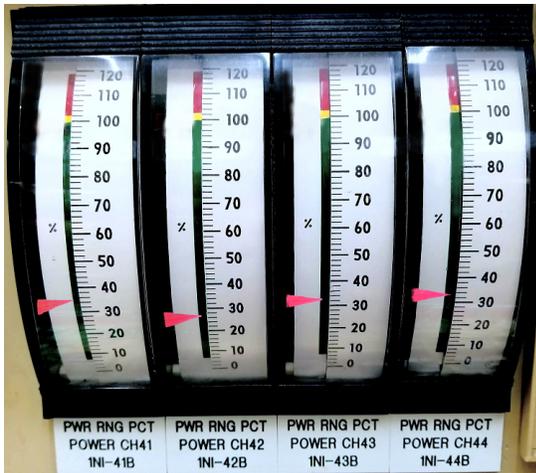
Case 1:



Case 2:



Case 3:



Case 4:



- A. 1 and 2 ONLY
- B. 2, 3 and 4
- C. 1 and 3 ONLY
- D. 4 ONLY

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 43

Reference Provided: No

History: New question for 2018 NRC exam RO Cog level: High

K/A-APE015 AA1.13 Importance RO 3.4 SRO 3.4

APE015 Reactor Coolant Pump (RCP) Malfunctions AA1.13 Ability to operate and/ or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Reactor power level indicators.

Tier: 1

Group: 1

K/A match: The question requires the examinee to monitor reactor power level during and loss of RC flow to determine plant response.

10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Answer explanation and plausibility:

Choice A is incorrect. In case 1, 3 PR NIs are indicating 27% power and the P8 permissive is not active. With P8 not active, the single loop loss of flow reactor trip is not active. This is plausible because power operations is not allowed with a single loop out of service.

Choice B is correct. In all three cases (2, 3, 4) at least 2 of the 4 PR NIs are greater than 30% power (the P-8 setpoint) and loss of a single loop of RCS flow would trigger an automatic reactor trip.

Choice C is incorrect. In case 1, 3 PR NIs are indicating 27% power and the P8 permissive is not active. With P8 not active, the single loop loss of flow reactor trip is not active. This is plausible because resetting the P-8 logic requires 3 of 4 PR NIs below 30% power and the logic is often confused.

Choice D is incorrect. Case 4 will result in an automatic reactor trip. 4 of 4 PR NIs are greater than 30% power (the P-8 setpoint) and loss of a single loop of RCS flow would trigger a reactor trip. This is plausible because case 4 is the only case with all 4 PR NIs greater than the 30% P-8 permissive.

Task No: R-RP-001

Learn Objective: S.RP2-04

Reference: Big note EF-1 rev 15

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

44

ID: RE0022-N1

Points: 1.00

Unit 2 is 100% power.

2BwOA PRI-1, EXCESSIVE PRIMARY PLANT LEAKAGE, step 9, CHECK IF THE LEAK IS ISOLATED, is in progress.

- Normal charging and Letdown have been isolated.
- PZR level is 52% and stable.
- Tave is 581°F and stable.
- VCT level is 18% and lowering 1% per minute.
- VCT make up cannot be established.

The US will direct the NSO to...

- A. reestablish normal letdown.
- B. align the CV pump suction to RWST.
- C. initiate Safety Injection.
- D. align emergency boration.

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 44

Reference Provided: No

History: New for 2018 NRC exam RO Cog level: High

K/A-E/APE 022AA2.01 Importance RO 3.2 SRO 3.8

E/APE 022 Loss of Reactor Coolant Makeup

AA2.01 Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Whether charging line leak exists.

Tier: 1

Group: 1

K/A match: Meets the K/A, the examinee needs to evaluate the CVCS lineup with the Loss of Reactor Coolant Makeup to determine an inventory loss is occurring in the charging header and is unisolable. The overall mitigative strategy is to isolate possible leak paths one at a time, the only exception would be charging which has letdown isolated as well because the regenerative heat exchanger needs charging in service before letdown can be established.

10CFR55.41(b)(8) Components, capacity, and functions of emergency systems.

Answer explanation and plausibility:

Choice A is incorrect and plausible. The normal letdown line is an input into the VCT and could stop the VCT level drop which makes this plausible; however, by placing normal letdown in service, the leak in the charging header will not be addressed and PZR level will begin dropping and force an SI to restore PZR level, therefore it is incorrect.

Choice B is correct. With VCT level lowering and PZR level and RCS temp stable, an unisolable loss of inventory in the charging header is indicated. The loss of make-up leaves the suction from the RWST as the only option for maintaining a suction source for the CV pumps. Per the OAS if VCT level cannot be maintained, swap suction to the RWST.

Choice C is incorrect, per the OAS PZR level is stable there is no need for safety injection at this time. Safety Injection will realign the suction of the charging pump to the RWST making this a plausible distractor.

Choice D is incorrect and plausible. Aligning the emergency boration line would be an alternate makeup to the suction of the CV pumps, making this plausible. Level dropping at 2% per minute equates to approximately 38 GPM and the emergency boration flow path is not used to keep up with the leakage, making this answer incorrect.

Task No: R-OA-031

Learn Objective: 3D.OA-76-A

Reference: 2BwOA PRI-1 rev 106

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

45

ID: BWLC3DOA9005

Points: 1.00

Unit 1 is in Mode 4:

- RCS temperature is 220°F.
- ALL RCS LSIVs are open.
- Reactor Coolant System is intact.
- RVLIS head & plenum levels are 100%.
- 1A RH pump is OOS.
- 1B RH train is in shutdown cooling.

1B RH pump trips and CANNOT be restarted.

The preferred method to restore Core Cooling will be by...

- A. steaming intact/non-isolated SGs.
- B. injecting the SI accumulators.
- C. SI pump hot leg injection.
- D. RCS bleed and feed.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 45

Reference Provided: NO

History: Bank question from BWD LORT bank RO Cog level: High

K/A-E/APE25AK1.01 -Importance RO 3.9 SRO 4.3

E/APE025 Loss of Residual Heat Removal System

EK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System:

A loss of RHRS during all modes of operation

Tier: 1

Group: 1

K/A match: Meets the K/A. This question requires the operator to know the operational implications of a loss of RH cooling in mode 4 and determine the appropriate mitigating strategy to restore core cooling.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is correct. With the RCS intact and the RC loops not isolated, steaming the SGs are the preferred means of maintaining core cooling during this loss of RH cooling.

Choice B is incorrect and plausible. Passive injection of the SI accumulators is an option per table A of 1BwOA PRI-10 making it plausible. However, to inject accumulators would require inventory to be removed from the RCS to make room, and steaming the SG is always the preferred option with the RCS intact.

Choice C is incorrect and plausible. SI hot leg injection is an option for injecting water from the RWST into the RCS. However, this option is only preferred if the RCS is adverse, which means a cold leg opening exists, the RX vessel head is in place and the loop stops valves are closed, making this incorrect.

Choice D is incorrect and plausible. RCS bleed and feed is a method of restoring cooling water from the RWST into the RCS making it plausible. However, this action would cause a loss of primary coolant as well, therefore would not be done while steam generators are available to cool.

Task No: R-OA-072

Learn Objective: 4D.OA-09

Reference: 1BwOA PRI-10 rev 109

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

46

ID: RE10026-N04

Points: 1.00

Regarding Component Cooling System valves that are also Containment Isolation Valves.

The reason some valves actuate on a Phase A signal while others actuate on a Phase B signal is because...

- A. it is desired to maintain containment penetration cooling during all accident conditions.
- B. excess letdown operations may be required to support Pressurizer level control during many accident conditions.
- C. it is desired to support and run RCPs during many accident conditions.
- D. not all system piping inside containment is Safety Category I.

Answer: C

Answer Explanation
<p>Question: 46 Reference Provided: No History: New question for 2018 NRC exam RO Cog level: Memory K/A-APE026 AK3.02 Importance RO 3.6 SRO 3.9 APE 026 Loss of Component Cooling Water AK3.02 Knowledge of the reasons for the following responses as they apply to the loss of Component Cooling Water: The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS. Tier: 1 Group: 1 K/A match: The question requires the examinee to know the reason for ESFAS actuation of the CCW system. 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.</p> <p>Answer explanation and plausibility: Choice A is incorrect. This is incorrect because containment penetration cooling isolates on Phase A and is not in kept in service. This is plausible since the purpose of penetration cooling is to prevent high temperature degradation of the concrete penetration of containment and thereby protect containment integrity. Choice B is incorrect. This is incorrect because excess letdown is isolated on Phase A. This is plausible since letdown operations is restored during some accidents to restore level control. Choice C is correct. This is correct because CCW is desired to maintain RCP support conditions. Forced coolant circulation is the most effective means of cooling the core and running RCPs will be protected if possible to protect forced circulation. Choice D is incorrect. This is incorrect because all CCW piping in containment is Safety Category 1. This is plausible because some systems inside containment are not Safety Category 1, like WO.</p> <p>Task No: R-OA-027 Learn Objective: S.RC2-08-D Reference: Big note CC-2 rev 7</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

47

ID: RE10038-N1

Points: 1.00

When the criteria are met to secure ECCS flow in 2BwEP-3, STEAM GENERATOR TUBE RUPTURE, why is it important to complete the steps in a timely manner?

- A. Slow the depletion of the RWST.
- B. Restore normal makeup to borate the RCS.
- C. Prevent introducing water into the main steam lines.
- D. Prevent damage to the RH pumps from overheating.

Answer: C

Answer Explanation
<p>Question: 47 Reference Provided: No History: New for 2018 NRC exam RO Cog level: Memory K/A-E/APE 038EK3.09 Importance RO 4.1 SRO 4.5 E/APE 038 Steam Generator Tube Rupture EK3.09 Knowledge of the reasons for the following responses as they apply to the SGTR: Criteria for securing/throttling ECCS. Tier: 1 Group: 1 K/A match: The examinee needs to have knowledge of the caution before step 19, "CHECK IF ECCS FLOW SHOULD BE TERMINATED", of 2BwEP 3 which states "ECCS flow must be terminated when termination criteria are satisfied to prevent overfilling the ruptured SG(s)." This is part of the overall mitigative strategy for 2BwEP-3. 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.</p> <p>Answer explanation and plausibility: Choice A is incorrect and plausible. This is incorrect because in _EP-3 terminating or reducing ECCS flow is done to prevent SG overfill per the caution prior to step 19. This is plausible because preventing depletion of the RWST is the reason for reducing ECCS flow to a single train in 2BwCA-1.1 and is a plausible reason for throttling or terminating ECCS flow. Choice B is incorrect and plausible. This is incorrect because in _EP-3 terminating or reducing ECCS flow is done to prevent SG overfill per the caution prior to step 19. This is plausible because normal make-up is established in _EP-3, step 26. Choice C is correct. This is correct because in _EP-3 terminating or reducing ECCS flow is done to prevent SG overfill per the caution prior to step 19. This is the overall mitigative strategy and purpose of _BwEP-3. Choice D is incorrect and plausible. This is incorrect because in _EP-3 terminating or reducing ECCS flow is done to prevent SG overfill per the caution prior to step 19. This is plausible because _EP-3 has steps to secure the RH pumps to prevent damage to them from overheating on recirc.</p> <p>Task No: R-EP-012 Learn Objective: 4D.EP-05 Reference: 2BwEP-3 rev 300</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

48

ID: RE10040-N02

Points: 1.00

Given the following conditions for BOTH units:

- A complete blowdown of one SG occurs from a steamline fault.
- SG levels began at the normal level control setpoint.
- All ESF actuation times are the same.
- AF flows and isolation times are the same.
- Nil ductility temperature (NDT) of each Rx Vessel is the same.

The resultant RCS cooldown would come CLOSEST to the vessel NDT on...

- A. Unit 1 from 0% power.
- B. Unit 1 from 100% power.
- C. Unit 2 from 0% power.
- D. Unit 2 from 100% power.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 48

Reference Provided: No

History: New question for the 2018 NRC exam RO **Cog level:** High

K/A-APE040 AK1.04 Importance RO 3.2 SRO 3.6

APE040A Steam Line Rupture AK1.04 Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: Nil ductility temperature.

Tier: 1

Group: 1

K/A match: This question requires the examinee to determine the implications of a steam line rupture in different situations as it applies to nil ductility temperature.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect because unit 1 SGs have less secondary mass than unit 2 SGs. This is plausible since unit 1 is a low power and the least amount of decay heat is available to maintain temperature above NDT and there is more SG mass at 0% power than at 100% power.

Choice B is incorrect. This is incorrect because unit 1 SGs have less secondary mass than unit 2 SGs and with the unit at 100% power, more decay heat will be present to limit the amount of cooldown. This is plausible since the SG will steam longer because there is more energy to release from the core that tripped from 100% power.

Choice C is correct. This is correct because the unit 2 SGs have more secondary mass to steam off than unit 1 SGs, allowing for more energy release and with the unit at 0% power there is less decay heat to maintain the primary temperature above NDT and there is more SG mass at 0% than 100%.

Choice D is incorrect. This is incorrect because with the unit at 100% power more decay heat will be present to limit the amount of cooldown. This is plausible because unit 2 has more secondary mass than unit 1 and the SG will steam longer because there is more energy to release from the core that tripped from 100% power.

Task No: R-EP-013

Learn Objective: 3D.EP-06-A

Reference: Big note SG-2 rev 9

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

49

ID: RE10054-N03

Points: 1.00

Unit 1 is in Mode 3 at NOP/NOT.

- The crew is preparing to withdraw the Control Bank Rods for a reactor startup.
- The Startup Feedwater pump is supplying feedwater to the SGs.

The Startup Feedwater pump trips on a ground fault.

- The crew enters 1BWOA SEC-1, SECONDARY PUMP TRIP, Attachment D, LOSS OF FW FLOW IN MODE 3.

1BWOA SEC-1 will establish feedwater flow by first starting...

- A. the standby CD/CB pump.
- B. the 1A MDFW pump.
- C. the 1A AF pump ONLY.
- D. BOTH the 1A AND 1B AF pumps.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 49

Reference Provided: No

History: New question for 2018 NRC exam

RO

Cog level: Memory

K/A-APE054 G2.4.9 Importance RO 3.8 SRO 4.2

054 Loss of Main Feedwater (MFW) G2.4.9 Knowledge of low power/ shutdown implication in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Tier: 1

Group: 1

K/A match: This question requires the examinee to identify the correct overall mitigative strategy for a low power loss of main feedwater.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect because the first action per 1BwOA SEC-1 Attachment D is to start the 1A AF pump. This is plausible because the start of the standby CD/CB pump is integral to recovering from a MFP trip at 100% power, and at low SG pressures in Mode 3 the CD/CB system is capable of feeding the SGs.

Choice B is incorrect. This is incorrect because the first action per 1BwOA SEC-1 Attachment D is to start the 1A AF pump. This is plausible because the MDFW pump would get an autostart on a feedpump trip at high power, and if it didn't autostart, this would be the first pump that is manually started to restore feedwater flow at high power.

Choice C is correct. This is correct because the first action per 1BwOA SEC-1 Attachment D is to start the 1A AF pump.

Choice D is incorrect. This is incorrect because the first action per 1BwOA SEC-1 Attachment D is to start the 1A AF pump. This is plausible because the auto start on lo-2 level would start both trains of AF, and the operators are starting the 1A AF pump to prevent the auto start signal from starting it.

Task No: R-OA-030

Learn Objective: 4D.OA-20

Reference: 1BwOA SEC-1 rev 108.

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

50

ID: RE10055-C01

Points: 1.00

Unit 2 was at 100% power.

- A grid disturbance caused a Loss of All AC Power.

The crew implemented 2BwCA-0.0, LOSS OF ALL AC POWER, and has transitioned to 2BwCA-0.1, LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED.

- Bus 241 is ENERGIZED via the ESF cross-tie.
- Bus 242 is DE-energized.

Which of the following would cause Natural Circulation flow on Unit 2 to RISE?

- A. Manually raising demand on the steam dump valves.
- B. Closing PZR Variable Heaters Group C Breaker.
- C. Establishing PZR Aux Spray flow.
- D. Raising AF flow to ALL SGs.

Answer: D

Answer Explanation
<p>Question: 50 Reference Provided: No History: Bank question from 2009 cert exam RO Cog level: High K/A-EPE055 EA2.02 Importance RO 4.4 SRO 4.6 EPE 055 Loss of Offsite and Onsite Power (Station Blackout) EA2.02 Ability to determine or interpret the following as they apply to a Station Blackout: RCS core cooling through natural circulation cooling to S/G cooling. Tier: 1 Group: 1 K/A match: This question requires the examinee to determine actions required to raise natural circulation cooling during a station blackout. 10CFR55.41(b)(14) Principles of heat transfer thermodynamics and fluid mechanics.</p> <p>Answer explanation and plausibility: Choice A is incorrect. This is incorrect because the steam dumps are not available (MSIVs are closed in 2BwCA-0.0). This is plausible since the steam dumps would raise steam flow and raise the delta T across the SG, raising natural circulation flow, if the MSIVs weren't closed. Choice B is incorrect. This is incorrect because PZR variable heaters are powered from bus 244 (not energized). This is plausible because it is one of the recovery steps in 2BwCA-0.1 (step 12). Choice C is incorrect. This is incorrect because lowering RCS pressure using Aux Spray will lower pressure in the RCS, but will not provide additional driving head for NC flow. This is plausible because it is one of the recovery steps in 2BwCA-0.1 (step 12). This can be ruled out using knowledge of fundamental heat transfer theory. Choice D is correct. Natural circulation flow can be enhanced by dumping steam from the SGs or raising AF flow, using knowledge of fundamental heat transfer theory.</p> <p>Task No: R-CA-001 Learn Objective: 3D.CA-09-B Reference: ILT LP I1-HT-XL-05, Natural Circulation rev 5 page 17/18</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

51

ID: RE10056-N04

Points: 1.00

Unit 1 is in MODE 3 preparing for reactor startup.

Unit 2 is at 100% power normally aligned.

- The 0A PW Make-Up Pump is in NAC supplying the PW ring header.
- The 0B PW Make-Up Pump is in NAT.

Unit 1 experiences a LOSS of offsite power.

- The crew entered 1BwOA ELEC-4, LOSS OF OFFSITE POWER.
- NO manual operator actions have been taken.
- ONE minute later, the **Unit 2 VCT** level lowered to the auto make-up setpoint.

The PW Make-up water to the **Unit 2 VCT** is available from the...

- A. 0A PW pump which re-started when the U-1 DG Output Breakers closed.
- B. 0A PW pump which re-started when Unit 2 VCT level lowered to the auto make-up setpoint.
- C. 0B PW pump which auto started when Unit 2 VCT level lowered to the auto make-up setpoint.
- D. 0B PW pump which auto started when the 0A PW pump lost power.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 51

Reference Provided: No

History: New question for 2018 NRC exam RO **Cog level:** High

K/A-APE056 AA2.76 Importance RO 2.6 SRO 2.6

APE056 Loss of Vital AC AA2.76 Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Reactor makeup water pump (running)

Tier: 1

Group: 1

K/A match: This question requires the examinee to interpret the loss of Offsite Power and the effect it has on Reactor makeup water pumps.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect because the 0A PW pump is powered from bus 143 to MCC133V1, which lost power on the loss of offsite power and requires manual action to reenergize bus 143. This is plausible since bus feed breakers are equipped with an auto swap feature. This function only allows for swapping between the SAT and UAT which is not online yet for unit 1.

Choice B is incorrect because the 0A PW pump is powered from bus 143 to MCC 133V1, which lost power on the loss of offsite power and requires manual action to reenergize bus 143. This is plausible since one of the start signals to the PW pumps comes from the MUX relays for make-up to the VCT.

Choice C is correct. Unit 2 RMCS will send an auto start signal to the 0B PW pump on demand for PW.

Choice D is incorrect. This is incorrect because the PW system does not have a low pressure start of the standby pump. This is plausible since many systems have a low pressure start of the standby pump including EH, GC, and CC.

Task No: R-OA-005

Learn Objective: 3C.AP-01-D

Reference: 20E-0-4030PW02 REV H

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

52

ID: RE10058-N02

Points: 1.00

Both units were at 100% power.

- Subsequently, 4KV bus 142 faults and its feed breaker trips.
- The crew has entered 1BWOA ELEC-3, LOSS OF 4KV ESF BUS.

TEN minutes later, an Equipment Operator reports the following indications at DC Bus 112:

- Battery Current - 50 amps
- Bus volts - 115 volts
- Charger Current - 0 amps

Per 1BWOA ELEC-3 the immediate mitigation strategy for DC bus 112 is to...

- A. close Battery Charger 112 AC input breaker.
- B. transfer DC bus 112 loads from "Normal" to "Reserve" power supplies.
- C. shed non-essential DC bus 112 loads.
- D. cross-tie DC bus 112 to DC bus 212.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 52

Reference Provided: No

History: Bank question 2011 NRC exam

RO

Cog level: High

K/A-APE058 AA1.03 Importance RO 3.1 SRO 3.3

058 Loss of DC Power AA1.03 Ability to operate and / or monitor the following as they apply to the Loss of DC Power: Vital and battery bus components

Tier: 1

Group: 1

K/A match: The question meets the K/A, requires examinee to monitor DC bus charger equipment and instrumentation during an event that would result in the loss of the DC bus if no action is taken.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

When bus 142 is faulted, AC input power to battery charger 112 is de-energized. The DC bus loads will now be powered by the battery vs. the battery charger. It is reasonable to assume that a faulted 4kv bus is not going to be re-energized for several hours at a minimum. Therefore, in order to maintain the unit at power until such time as a controlled shutdown may be required, the crew must cross tie DC bus 112 to 212. 1BWOA ELEC-3 directs this action.

Choice A is incorrect, attempting to re-energize the battery charger will not work since its ultimate power supply (bus 142 via bus 132X) is de-energized. The indication of 0 amps on the battery charger indicates the charger is deenergized and it is not powered directly from bus 142. An examinee could incorrectly assume bus 142 fault does not impact the charger because it is powered from 480V bus 132X, making this plausible.

Choice B is incorrect, transferring DC loads to "reserve" feeds does not change the DC bus that they are powered from. Reserve feeds are simply alternate breakers and cabling that are still fed from the same DC bus as "normal" feeds. This is plausible since transfer loads to reserve power implies a different source like the unit 0 CC pump. This is not true of DC loads though.

Choice C is incorrect. load shedding is a contingency action that is taken when there is no alternative power supply to the DC bus. It is plausible to shed loads which would limit the draw down of the battery and extend its longevity. These actions are directed if there is no alternative power supply.

Choice D is correct. 1BWOA ELEC-3 directs the crew to cross tie DC bus 112 with 212 which will place both bus loads on battery charger 212.

Task No: R-OA-131

Learn Objective: 4D.OA-131.

Reference: 1BWOA ELEC-3 rev 102

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

53

ID: RE10062-N03

Points: 1.00

Unit 1 was at 100% power.

- 1B and 1D RCFCs are S/D in standby.
- 1SX016B, 1B & 1D SX Inlet VLV, & 1SX027B, 1B & 1D SX outlet VLV, are both energized and closed.

The following then occurred:

- A unit 1 Rx trip and SI occurs.
- The NSO is checking RCFC alignments and notes the 1SX027B failed to open.

Which of the following equipment malfunctions would cause the 1SX027B to remain closed?

- A. 1SX016B failed to open.
- B. 1B SX pump failed to start.
- C. 1B and 1D RCFC fan breakers failed to close.
- D. 1SX147B, "CNMT Chiller 1B Byp Isol VLV," failed to open.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 53

Reference Provided: No

History: New question for 2018 NRC exam

RO

Cog level: Memory

K/A-APE062 AK3.02 Importance RO 3.6 SRO 3.9

APE062 Loss of Nuclear Service Water AK3.02 Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS.

Tier: 1

Group: 1

K/A match: The examinee must have knowledge of automatic alignments that occur due to ESFAS actuation and which actuation signal cause the failure of the SX component to realign, specifically the actuation of the SI causes the slow speed RCFC breakers to close which sends the open signal to the 1SX027B.

10CFR55.41(b)(4) Secondary coolant and auxiliary systems that affect the facility.

Answer explanation and plausibility:

Choice A is incorrect since the 1SX027B only has auto open feature from a closure of either B train RCFCs high or low speed breaker. This is plausible since both the 1SX016B and 1SX027B are part of the SX pump manual starting interlock and are frequently checked open in a pair prior to manually starting an SX pump.

Choice B is incorrect. This is incorrect because the failure of the 1B SX pump to start on ESFAS actuation does not depend on nor have an effect on the 1SX027B. This is plausible since the 1SX027B is required to be open to manually start the 1B SX pump.

Choice C is correct. There are 3 signals that open the 1SX027B: the control switch to open, either the 1B RCFC high speed or low speed breaker closed, or the 1D RCFC high or low speed breaker closed. The low speed breakers get a closed signal from the ESFAS actuation.

Choice D is incorrect, since the 1SX147B does not have any interlocks that prevent the 1SX027B from opening. The 1SX147B will get the open signal from K610 which is the same actuation relay as the RCFC low speed breaker, which makes this plausible. This valve is verified in its correct position in 1BwEP-0 step 7 RNO which also verifies the 1SX027B open.

Task No: 3D.EP-19-A

Learn Objective: R-EP-037

Reference: 20E-1-4030SX11 rev H

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

54

ID: RS10078-N01

Points: 1.00

The U-1 SAC has just been made available (clearance order recently cleared).

The crew is preparing to start the U-1 SAC per BwOP SA-1, STARTUP AND OPERATION OF STATION AIR COMPRESSORS.

- The U-0 and the U-2 SACs are NOT available.
- Nitrogen has been aligned to the Unit 1 SAC control panel.
- The equipment operator reports that Inlet oil temperature is 58°F.

The U1 SAC can...

- A. be started in the MCR or locally AFTER the EO runs the lube oil pump to raise oil temperature.
- B. ONLY be started from the MCR AFTER the EO runs the lube oil pump to raise oil temperature.
- C. be started IMMEDIATELY, locally or in the MCR.
- D. be started IMMEDIATELY, locally ONLY.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 54

Reference Provided: No

History: Modified Bank question 2013 NRC exam

RO

Cog level: Memory

K/A-APE065 G2.4.35 Importance RO 3.8 SRO 4.0

APE065 Loss of Instrument Air G2.4.35 Knowledge of local auxiliary operator tasks during an emergency plan and the resultant operational effects.

Tier: 1

Group: 1

K/A match: The question requires examinee knowledge of the local actions required to ensure the SAC can be started to restore IA to the station.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facilit.

Answer explanation and plausibility:

Choice A is correct. Per BwOP SA-1 Limitation E.6 lube oil temperature below 74°F will prevent the start of the SAC. Either Attachment A of BwOP SA-1 must be used to lower the lube oil temperature setpoint, or the aux oil pump must be run until temperature rises above 74°F.

Choice B is incorrect because the SAC lube oil temperature of 58°F will not allow the SAC to start from either location. This is plausible since the procedure allows for either local or remote start. BwOP SA-1 will allow you to attempt to start the SAC from the MCR without checking oil temperature, making this plausible. There is a note in the procedure that clearly states it will not start on low oil temperature and further action is required in the field.

Choice C is incorrect because the SAC lube oil temperature of 58°F will not allow the SAC to start from either location. This is plausible since the procedure allows for either local or remote start. The examinee could conclude that either option will work without further action since the setup steps are skipped.

Choice D is incorrect because the SAC lube oil temperature of 58°F will not allow the SAC to start from either location. This is plausible since the procedure allows for either local or remote start. It is plausible that since the actions required are in the field that a local start is possible but without having met the oil temperature requirement first, the SAC will not start.

Task No: R-OA-009

Learn Objective: S.SA1-02-A

Reference: BwOP SA-1 rev 43

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

55

ID: RE10077-C03

Points: 1.00

Unit 1 is at 100% power.

- 1VI-MP006, MAIN GENERATOR 1 OUTPUT VARS, indicates 200 MVARs out.

A grid disturbance results in grid voltage rising.

- TSO reports NO lines were lost and total grid reactive load did NOT change.

Unit 1 MVARs out __ (1) __.

The crew will take the __ (2) __ to return MVAR load to a STABLE 200 MVARs OUT.

- A. (1) rose
(2) VOLT adjuster control switch to LOWER
- B. (1) lowered
(2) VOLT adjuster control switch to RAISE
- C. (1) rose
(2) BASE adjuster control switch to LOWER
- D. (1) lowered
(2) BASE adjuster control switch to RAISE

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 55

Reference Provided: No

History: Bank question 2013 NRC exam

RO

Cog level: High

K/A-E/APE 077 AK2.07 Importance RO 3.6 SRO 3.7

E/APE 077 Generator Voltage and Electric Grid Disturbances AK2.07 Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine/ generator control.

Tier: 1

Group: 1

K/A match: The question requires the examinee to assess the effect of rising grid voltage (disturbance) on main generator reactive load and determine the appropriate voltage controller adjustment.

10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Answer explanation and plausibility:

Choice A is incorrect. This is incorrect since a rise in grid voltage would bring grid voltage and the generator voltage closer together resulting in lower MVARs on the generator. Reactive load is commonly misunderstood, since the grid voltage rose and 1 generator is small in comparison to the grid it could be misinterpreted that the generator would move in the same direction as the grid. Making this plausible.

Choice B is correct. A rise in grid voltage with no change in total reactive load causes a reduction in reactive load output of the main generator, therefore MVARs out will lower. To restore reactive load to the original value, the operator must go to RAISE on the voltage regulator control switch. The base adjuster will have no effect with the voltage regulator in automatic. Control is shifted to the Base Adjuster if/when the automatic voltage regulator is taken to "off".

Choice C is incorrect, the voltage regulator in automatic maintains the generator output voltage stable not the reactive load. This is plausible since the voltage regulator being in auto will cause the regulator to adjust to the grid disturbance and MVAR loading is commonly misunderstood.

Choice D is incorrect. The base adjuster is used to operate the generator voltage control in manual. With control in automatic, the voltage regulator will adjust for any moves made in the base adjuster. This plausible since a failure of the automatic voltage regulator would require the operator to place the voltage regulator in manual. However, the stem does not indicate any issues with the voltage regulator.

Task No: R-OA-048

Learn Objective: S.MP2-05-A

Reference: BwOP MP-23 rev. 17

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

56

ID: RE1WE04-C01

Points: 1.00

Unit 1 was at 100% power.

- A LOCA outside containment has occurred.
- The crew has implemented 1BwCA-1.2, LOCA OUTSIDE CONTAINMENT.

What is the MOST probable location of the LOCA outside containment?

- A. PS system suction piping.
- B. CV system discharge piping.
- C. RH system suction piping.
- D. SI system discharge piping.

Answer: C

Answer Explanation

Question: 56

Reference Provided: No

History: Bank question from past Braidwood cert exam RO **Cog level:** Memory

K/A- WE04 G 2.1.32 **Importance** RO 3.8 SRO 4.0

WE04 LOCA Outside Containment G 2.1.32 Ability to explain and apply system limits and precautions.

Tier: 1

Group: 1

K/A match: The question requires examinee knowledge of the system limits from the normal operation of the RH system and combine that knowledge with the general mitigative strategies of 1BwCA-1.2 to determine the most common failure modes for a LOCA outside containment.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. The PS (Primary Sample) system is a normally isolated system with small piping diameter and not likely to be a source of leakage outside containment. This is plausible because the PS system is in communication with multiple points in the RCS and brings those system fluids outside containment.

Choice B is incorrect, the letdown flow path would be isolated and CV system injection piping is class I and is rated for RCS pressure. This is plausible because the CV system has a direct connection to the RCS.

Choice C is correct. From WOG background document for CA-1.2, the RH suction path is the dominant probability of LOCA outside containment from the simultaneous failure of two MOVs.

Choice D is incorrect, the flow path would be isolated and SI system piping is class I and is rated for RCS pressure. This is a plausible distractor as the SI system is directly tied to the RCS.

Task No: R-CA-004

Learn Objective: T.CA2-03

Reference: 1BwCA-1.2 rev 300 and Background document

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

57

ID: RE10003-N01

Points: 1.00

Unit 1 was at 95% power, ramping to 100%.

During an automatic rod step, Control Bank D, Rod M-12 drops part way into the core.

- Rod M-12 DRPI now indicates 120 steps.
- All other CB-D DRPI indications are 198 steps.
- DRPI ACCURACY MODE SWITCH is in A+B position.

The crew enters 1BwOA ROD-3, DROPPED OR MISALIGNED ROD, and determines that the failure occurred in rod control, but NOT DRPI.

The LOWEST possible ACTUAL position of Rod M-12 is _____ steps.

- A. 108
- B. 110
- C. 116
- D. 120

Answer: C

Answer Explanation

Question: 57

Reference Provided: No

History: NRC Exam 2009 Question # 57

RO Cog level: High

K/A-APE003 G2.1.27 Importance RO 3.9 SRO 4.0

APE0003 Dropped Control Rod G2.1.27 Knowledge of system purpose and/or function.

Tier: 1

Group: 2

K/A match: The question requires the examinee to know that DPRIs purpose is to provide redundant rod position indication to the step counters and limits of how it functions to locate a rod based on its design as related to a dropped control rod.

10CFR55.41(b)(6) Design, components, and functions of reactivity control mechanisms and instrumentation.

Answer explanation and plausibility:

Choice A is incorrect, normal DRPI accuracy is +/- 4 steps in A + B. This plausible because Tech Spec LCO requires ± 12 steps rod position.

Choice B is incorrect, normal DRPI accuracy is +/- 4 steps in A + B. It is plausible DRPI is accurate +4 steps and -10 steps when it is in A- mode of operation indicating the rod could be 10 steps below..

Choice C is correct. DRPI position accuracy is normally ± 4 steps (3 steps + 1 step for manufacturing tolerances) when in the A+B mode.

Choice D is incorrect, normal DRPI accuracy is +/- 4 steps in A + B. This is plausible because the rod could be stuck 4 steps above 120 steps which would be the highest position the rod could be in and indicate 120 steps.

Task No: R-PI-001

Learn Objective: 3C.PI-01-A

Reference: Big Note RD-6 rev 7

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

58

ID: RE10005-N01

Points: 1.00

Unit 1 is ramping down from full power.

- Rod Control is in auto.

At approximately 85% power:

- DRPI indication for rod D-4, in CBD is 192 steps
- All other rods in CBD indicate 176 steps on their group step counter.

The ramp is stopped and rod control is placed in Manual.

The DRPI indication can be verified by QPTR being (1) and delta I being (2) in the affected quadrant.

- A. (1) lower
(2) lower
- B. (1) lower
(2) higher
- C. (1) higher
(2) lower
- D. (1) higher
(2) higher

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 58

Reference Provided: No

History: New question for 2018 NRC exam RO Cog level: High

K/A-APE0005 AA2.01 Importance RO 3.3 SRO 4.1

APE0005 Inoperable/ Stuck Control Rod AA2.01 Ability to determine and interpret the following as they apply to the Inoperable/ Stuck Control Rod: Stuck or inoperable rod from in-core and ex-core NIS, in-core or loop temperature measurements.

Tier: 1

Group: 2

K/A match: The question requires the examinee to determine the indications of the stuck rod from ex-core NIS.

10CFR55.41(b)(6) Design, components, and functions of reactivity control mechanisms and instrumentation.

Answer explanation and plausibility:

Choice A is incorrect, with the rest of the control bank inserted as much as 20 steps deeper, flux will be higher in the quadrant the stuck rod is in. This is plausible because the majority of the rods are inserted and the flux profile will be lower than the full power profile.

Choice B is incorrect, with the rest of the control bank inserted as much as 20 steps deeper, flux will be higher in the quadrant the stuck rod is in. This is plausible because the majority of the rods are inserted more and the flux profile will be lower than the full power profile and the stuck rod will cause AFD to be higher in the affected quadrant.

Choice C is incorrect, with the rest of the control bank inserted as much as 20 steps deeper, flux will be higher in the quadrant the stuck rod is in. This is plausible since the QPTR will be higher in the quadrant with the stuck rod.

Choice D is correct, with the rest of the control bank inserted as much as 20 steps deeper flux will be higher in the quadrant the stuck rod is in.

Task No: R-OA-067

Learn Objective: A.PD2-02-E

Reference: I1-PD-XL-02 rev 4

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

59

ID: RE20032-C01

Points: 1.00

Unit 2 is in Mode 3.

- ALL shutdown banks are withdrawn.
- SR channel N32 is stable at 200 cps.

SR channel N31 was responding erratically and the crew implemented 2BwOA INST-1, NUCLEAR INSTRUMENTATION MALFUNCTION, with the following conditions:

- SR channel N31 level trip bypass switch is in BYPASS.
- SR channel N31 high flux at shutdown switch is in BLOCK.
- BOTH IR channels indicate 1×10^{-11} amps.

Then, SR channel N31 CONTROL POWER fuses BLOW.

What is the status of the shutdown banks and what action is required?

- A. Shutdown banks are tripped.
Perform actions per 2BwEP-0, REACTOR TRIP OR SAFETY INJECTION RESPONSE.
- B. Shutdown banks are tripped.
Perform actions of 2BwGP 100-2A1, Attachment A, CONTINGENCY FOR SUSPENDED REACTOR STARTUP.
- C. Shutdown banks are withdrawn.
Perform remaining actions of 2BwOA INST-1.
- D. Shutdown banks are withdrawn.
Perform actions of 2BwGP 100-2, PLANT STARTUP.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 59

Reference Provided: No

History: Bank question from 2013 cert exam RO Cog level: High

K/A- APE032 AK2.01 **Importance** RO 2.7 SRO 3.1

APE032 Loss of Source Range Nuclear Instrumentation AK2.01 Knowledge of the interrelations between Loss of Source Range Nuclear Instrumentation and the following: Power supplies, including proper switch positions.

Tier: 1

Group: 2

K/A match: The question requires the examinee to understand the interrelations of control and instrument power and the bypass and block switches associated with the Source Range Nuclear Instrumentation.

10CFR55.41(b)(2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Answer explanation and plausibility:

Choice A is correct. A loss of control power will allow the source range trips to actuate and trip the reactor.

Choice B is incorrect, a loss of control power will allow the source range trip to actuate and trip the reactor. This is plausible since loss of control power will allow the source range trips to actuate and trip the reactor, and the attachment A is used for contingency to respond to a startup that is suspend.

Choice C is incorrect, a loss of control power will allow the source range trip to actuate and trip the reactor. This is plausible since a loss of instrument power will not cause a reactor trip, and the 1BWOA INST-1 procedure has the actions to deal with a loss of nuclear instrumentation, but all actions for the failure where completed, and the procedure will not deal with the reactor trip response.

Choice D is incorrect, a loss of control power will allow the source range trip to actuate and trip the reactor. This is plausible since a loss of instrument power will not cause a reactor trip with the level bypass switch in BYPASS, and the Tech Spec required action for a single source range inoperable is to manually insert the control rods and then open the reactor trip breakers.

Task No: R-OA-053

Learn Objective: S.NI1-08A/B

Reference: Big Note NI-4 rev 11

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

60

ID: RS20034-N02

Points: 1.00

The 0AR039J, FUEL HANDLING BLDG CRANE, area rad monitor has failed HIGH.

Other than MPC transfers, this failure will prevent...

- A. SPENT fuel moves within the spent fuel pool.
- B. SPENT fuel transfer from the fuel transfer cart to the spent fuel pool.
- C. NEW fuel transfer from the spent fuel pool to the fuel transfer cart.
- D. NEW fuel transfer from shipping containers to the spent fuel pool.

Answer: D

Answer Explanation

Question: 60

Reference Provided: No

History: bank question from 2011 NRC exam **RO Cog level:** Memory

K/A APE036 K3.02 **Importance** RO 2.9 SRO 3.6

APE036 Fueling handling incidents K3.02 Knowledge of the reasons for the following responses as they apply to Fuel Handling Incidents: Interlocks associated with fuel handling equipment.

Tier: 1

Group: 1

K/A match: This question test knowledge of the purpose of the Fuel Handling interlocks associated with the radiation monitoring system. The 0AR039J stops upward travel on the fuel handling building crane.

10CFR55.43(b)(7) fuel handling facilities and procedures.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because most tech spec requirements associated with spent fuel pool require the suspension of movement of fuel; spent fuel is more likely to cause a high rad condition in the fuel handling building. The 0AR39J high alarm prevents upward motion of the fuel handling building crane which is not used for spent fuel moves other than dry cask MPC transfer.

Choice B is incorrect. This is plausible because most tech spec requirements associated with spent fuel pool require the suspension of movement of fuel. The 0AR39J high alarm prevents upward motion of the fuel handling building crane which is not used for spent fuel moves other than dry cask MPC transfer.

Choice C is incorrect. This is plausible because the new fuel elevator has interlocks that would prevent motion upward if a fuel assembly was loaded in the elevator. The interlock is not associated with 0AR039J which is why this is incorrect.

Choice D is correct. 0AR039J failing high will prevent upward motion of the fuel handling bldg crane hoist. This crane is only used for new fuel transfer from shipping containers to the pool or transferring spent fuel already in casks.

Task No: R-FH-007

Learn Objective: 3C.FH-04-A

Reference: Bwd ILT LP I1-FH-XL-01 rev.5c

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

61

ID: RE10037-N02

Points: 1.00

Unit 1 was at 100% power.

45 minutes ago, a SG tube leak occurred.

- The crew implemented 1BwOA SEC-8, STEAM GENERATOR TUBE LEAK.
- The crew initiated a Unit Shutdown.

Reactor power is currently at 75%.

- The crew is monitoring RCS subcooling per the OAS of 1BwOA SEC-8.

CETC are 640°F.

- RCS Pressure is 2040 psig.

The crew will...

- A. STOP the Unit ramp down and maintain RCS temperature stable.
- B. CONTINUE with the Unit ramp down and shutdown.
- C. TRIP the reactor and initiate a RCS cooldown.
- D. TRIP the reactor and actuate SI.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 61

Reference Provided: Steam Tables

History: New question for 2018 NRC exam RO Cog level: High

K/A-APE 037 AK1.01 Importance RO 2.9 SRO 3.3

APE 037 Steam Generator Tube Leak AK1.02 Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak. Use of steam tables.

Tier: 1

Group: 2

K/A match: The question requires the examinee to evaluate the status of subcooling in the reactor using the steam tables to determine adequate subcooling is not available.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect, with little to no subcooling available, the procedurally directed action is to trip the reactor and initiate safety injection. It is plausible that with a large enough tube leak that a higher shutdown rate could cause pressure to lower quicker than desired and stopping the cool down would allow pressure to restore.

Choice B is incorrect, with little to no subcooling available, the procedurally directed action is to trip the reactor and initiate safety injection. It is plausible that from 75% power, you would shutdown the reactor and stabilize pressure then commence a cooldown and depressurization per the normal procedure as directed in 1BwOA SEC-8 step 10e.

Choice C is incorrect, with little to no subcooling available, the procedurally directed action is to trip the reactor and initiate safety injection. It is plausible that from 75% power, you would trip the reactor as part of a shutdown per the GP. Then stabilize pressure and commence a cooldown and depressurization per the 1BwOA SEC-8.

Choice D is correct, with little to no subcooling available, 1BwOA SEC-8 OAS requires a reactor trip and safety injection.

Task No: R-OA-003

Learn Objective: 4D.OA-17

Reference: 1BwOA SEC-8 rev 109

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

62

ID: RS20072-N02

Points: 1.00

Unit 1 is in MODE 4 after a normal shutdown and cooldown.

- RCS pressure was 350 psig.

Subsequently, a Containment Area rad monitor alarms and the crew notes the following:

- PZR Level is DROPPING.
- Letdown is ISOLATED.
- Charging flow is 200 gpm.

What procedure will the crew enter to mitigate these conditions?

- A. 1BWOA PRI-1, EXCESSIVE PRIMARY PLANT LEAKAGE.
- B. 1BWOA S/D-2, SHUTDOWN LOCA.
- C. 1BWOA PRI-4, HIGH REACTOR COOLANT ACTIVITY.
- D. 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 62

Reference Provided: No

History: Bank question from 2009 NRC exam RO Cog level: High

K/A-APE061 G2.2.44 Importance RO 4.2 SRO 4.4

APE061 Area Radiation Monitoring System Alarms G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Tier: 1

Group: 1

K/A match: The question requires the examinee to interpret the area rad monitor alarm and verify the status by understanding the control room indications to ensure the correct actions and mitigative strategy is employed.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect, 1BWOA PRI-1 is written for leaks in Mode 1-3 with SI accumulators not isolated. This is plausible since 1BWOA PRI-1 is written to address primary leaks in mode 1-3 and the indications are indicative of a primary leak.

Choice B is correct, 1BWOA S/D 2 is written for modes 3 and 4 after SI accumulators are isolated. Mitigation steps of 1BWOA S/D-2 will be to manually restore and align ECCS, then cool down the plant and depressurize the RCS.

Choice C is incorrect, 1BWOA PRI-4 is applicable for the gross failed fuel monitor alarm, not a containment area monitor. This is plausible since the high radiation levels in area monitors can be seen with failed fuel. However, high RCS activity is indicated by gross failed fuel monitors.

Choice D is incorrect, 1BWEPI-1 would not be the next procedure entry because it is not a direct entry procedure. This is plausible because 1BWEPI-0 will not be an entry point since the reactor is already tripped and SI blocked. 1BWEPI-1 does have actions for dealing with RCS leakage for accidents from power.

Task No: R-OA-111

Learn Objective: 4D.OA-77

Reference: 1BWOA S/D-2 rev 108

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

63

ID: RE2WE15-N01

Points: 1.00

An RCS LOCA occurred on Unit 2 from full power.

- A reactor trip and SI occurred.
- Containment pressure peaked above the CS actuation setpoint and is currently 1.9 psig and stable.
- All ECCS equipment operated as designed and the plant is in the proper alignment for ECCS Recirculation phase.
- Due to containment high water level, the crew has just entered 2BwFR-Z.2, RESPOND TO CONTAINMENT FLOODING UNIT 2.

Which of the following systems is the cause of the unexpected water level in containment?

- A. CC
- B. CS
- C. RH
- D. SX

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 63

Reference Provided: No

History: Modified question 2014 NRC exam

RO

Cog level: Memory

K/A-E15EA2.2 Importance RO 2.9 SRO 3.3

E15 Containment Flooding EA2.2 Ability to determine and interpret the following as they apply to the (Containment Flooding): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Tier: 1

Group: 2

K/A match: This question requires the examinee to determine the overall mitigative strategy for 1BwFR-Z.2 and the source of water for flooding in containment that is not accounted for in the design limitations of containment.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect, the CC system is isolated from containment on the phase B actuation. This is plausible as current pressure is only 1.9 PSIG and phase B is not required at that pressure. When CS actuated, Phase B was actuated as well.

Choice B is incorrect, the design basis of containment flood level takes into account the entire contents of the RCS, RWST, Spray add tank and SI accumulators. CS provides a large volume of water from the RWST and is responsible for the rapid transfer of water from the RWST to containment and would therefore be a plausible source.

Choice C is incorrect, the design basis of containment flood level takes into account the entire contents of the RCS, RWST, Spray add tank and SI accumulators. The RH pumps provide a large volume of water from the RWST, and therefore it is plausible that this could be a source.

Choice D is correct. SX flow to the RCFCs is not included as part of the design basis analysis for containment flooding. This system has the ability to transfer the contents of the cooling pond to containment if there was a large enough leak present.

Task No: R-FR-031

Learn Objective: 4D.FR-14

Reference: 2BwFR-Z.2 page 2 rev 300

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

64

ID: RWE16-N01

Points: 1.00

Unit 1 experienced a reactor trip from full power.

- The STA reports a YELLOW path status tree on Containment Radiation still exists.
- The US enters 1BwFR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL.
- The TSC has been contacted to evaluate performance of the procedure steps in 1BwFR-Z.3.
- While waiting for the TSC recommendations, the US is briefing the crew on 1BwFR-Z.3 steps.

The mitigation strategy of 1BwFR-Z.3 includes operation of the...

- A. Containment Spray System.
- B. Containment Mini-Purge System.
- C. Containment Post LOCA Purge System.
- D. Reactor Containment Fan Cooler System.

Answer: C

Answer Explanation
<p>Question: 64 Reference Provided: No History: Bank question from 2014 NRC exam RO Cog level: Memory K/A-E16 EK2.1 Importance RO 3.0 SRO 3.3 E16 High Containment Radiation K2.1 Knowledge of the interrelations between the (High Containment Radiation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. Tier: 1 Group: 2 K/A match: The question requires the examinee to determine the system manual features utilized during response to High Containment Radiation Levels. 10CFR55.41(b)(12) Radiological safety principles and procedures.</p> <p>Answer explanation and plausibility: Choice A is incorrect, 1BwFR-Z.3 uses the post LOCA purge system at the TSC's discretion. This is plausible since the containment spray system is part of the primary containment response system. Choice B is incorrect, 1BwFR-Z.3 uses the post LOCA purge system at the TSC's discretion. The Cnmt mini-purge system is the normal system through which containment atmosphere is released or diluted from, making this a plausible distractor. Choice C is correct. 1BwFR-Z.3 uses the post LOCA purge system at the TSC's discretion. Choice D is incorrect, 1BwFR-Z.3 uses the post LOCA purge system at the TSC's discretion. This is plausible since the RCFCs are part of the primary containment response system.</p> <p>Task No: R-FR-032 Learn Objective: 3D.FR-15-A Reference: 1BwFR-Z.3 page 2 rev 300</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

65

ID: RE2WE03-N02

Points: 1.00

Unit 1 was at 100% power.

- An RCS LOCA has occurred.
- Reactor Trip Breaker "RTA" failed to open (automatically OR manually from MCB switches).

The operating crew is performing 1BwEP ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, step 6, CHECK IF RH PUMPS SHOULD BE STOPPED, with the following conditions:

- Both SI Reset Pushbuttons have been depressed at 1PM06J.
- RCS pressure is currently 1100 psig.
- SI actuated is fast flashing.
- SI blocked is fast flashing.
- Containment pressure is 3.1 psig.

What are the minimum actions that must be taken to place the 1A RH pump in **STANDBY**?

- A. Place the 1A RH pump C/S in PTL.
- B. Place the 1A RH pump C/S in NAT.
- C. Locally open RTA, then momentarily depress train A SI reset pushbutton again, and place the 1A RH pump C/S in PTL.
- D. Locally open RTA, then momentarily depress train A SI reset pushbutton again, and place the 1A RH pump C/S in NAT.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 65

Reference Provided: No

History: Modified from Bank question 2011 cert exam RO **Cog level:** High

K/A-E03 EA1.1 Importance RO 4.0 SRO 4.0

E03 LOCA Cooldown- Depressurization EA1.1 Ability to operate and/ or monitor the following as they apply to the (LOCA Cooldown and Depressurization) Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic manual features.

Tier: 1

Group: 2

K/A match: This question requires the candidate to understand the automatic functions associated with RH and the SI signal reset interlocks to allow the RH pumps to be placed in standby and continue the post LOCA cooldown.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect, placing the 1A RH pump in in PTL will not allow the pump to work in standby for an auto start as required. This is plausible as this action will stop the RH pump, but it will not place the pump in standby with auto start ability.

Choice B is incorrect, train A SI is still active and 1A RH pump will auto start in NAT until the train A SI is reset. This plausible is plausible because the SI reset pushbuttons have been depressed and NAT is the standby position. However as stated above, the 1A RH pump will automatically start.

Choice C is incorrect, placing the 1A RH pump in in PTL will not allow the pump to work in standby for an auto start as required. This is plausible because the action to rack out RTA and depress the train A SI reset pushbutton will reset the SI signal, but placing the pump in PTL will stop the RH pump; it will not place the pump in standby with auto start ability as directed.

Choice D is correct. RTA being closed will prevent auto SI from being reset and blocked when the SI reset pushbuttons are depressed. Locally racking out RTA causes the breaker to open allowing the SI signal to be reset and blocked in order to place the RH pump NAT, the standby position.

Task No: R-EP-003

Learn Objective: 3D.EP-13-A

Reference: BIG NOTE EF-4 rev 5

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

66

ID: RG10018-N01

Points: 1.00

In accordance with OP-AA-111-101, OPERATING NARRATIVE LOGS AND RECORDS, if an erroneous log entry is on a hand written or printed record, the correction should be made by _____ and dating and initialing the entry.

- A. completely blacking out the erroneous entry and inserting corrected information
- B. drawing a single line through the erroneous entry and inserting corrected information
- C. discarding the erroneous record and recreating a correct record
- D. writing "DELETE" in the left-hand column next to the erroneous entry and inserting corrected information

Answer: B

Answer Explanation

Question: 66

Reference Provided: No

History: Bank question from 2009 NRC exam RO **Cog level:** Memory

K/A- G2.1.18 **Importance** RO 3.6 SRO 3.8

Ability to make accurate, clear, and concise logs, records, status boards, and reports.

Tier: 3

Group: NA

K/A match: The question requires the examinee to understand the requirements for ensuring erroneous log entries are correct and clear and concise.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect. Step 4.1.10 of OP-AA-111-101 states, "draw a single line through the erroneous entry". This is plausible since it is not desired for the erroneous information to be mistaken for the correct log information.

Choice B is correct. Step 4.1.10 of OP-AA-111-101 states, "draw a single line through the erroneous entry".

Choice C is incorrect. Step 4.1.10 of OP-AA-111-101 states, "draw a single line through the erroneous entry". It is common practice for non-official documents to discard the mistake and start from scratch making this plausible.

Choice D is incorrect. Step 4.1.10 of OP-AA-111-101 states, "draw a single line through the erroneous entry". Electronic logs have become the official record and delete would indicate information should be discarded or not used, making this plausible.

Task No: R-AM-006

Learn Objective: 3E.AM-038-A

Reference: OP-AA-111-101 rev 13

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

67

ID: RS10064-N03

Points: 1.00

The following is partial data from BwOP DG-11T2, DIESEL GENERATOR OPERATING LOG, for 1A DG during full load stable conditions:

<u>Cylinder Number</u>	<u>Temp @ Time 1</u>	<u>Temp @ Time 2</u>	<u>Temp @ Time 3</u>
1 Left	820°F	840°F	830°F
2 Left	920°F	915°F	905°F
3 Left	900°F	950°F	1035°F

These readings indicate...

- A. there may be serious material overheating.
- B. a serious cylinder imbalance may exist.
- C. a valve train or fuel delivery failure is likely.
- D. the exhaust stack may be blocked.

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 67

Reference Provided: BwOP DG-11T2

History: Bank question 2013 NRC exam

RO

Cog level: High

K/A- 2.1.25 **Importance** RO 3.9 SRO 4.2

2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.

Tier: 3

Group: NA

K/A match: This question requires the examinee to interpret the provided data using the table provided in BwOP DG-11T2 as a reference and determine the correct course of action.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect. Per the caution on page 3 of BwOP DG-11T2, a maximum exhaust temperature of 1100F may indicate serious material overheating and that condition does not exist. This is plausible as it is one of the indications being monitored per the caution.

Choice B is correct. Per the caution on page 3 of BwOP DG-11T2, at time 3 the exhaust temperature difference greater than 200F is indicative of a serious cylinder imbalance.

Choice C is incorrect. Per the caution on page 3 of BwOP DG-11T2, a DECREASE of 160F indicates a valve train or fuel delivery failure and the biggest temperature drop is 70F. This is plausible as it is one of the indications being monitored per the caution.

Choice D is incorrect. Per the caution on page 3 of BwOP DG-11T2, a high turbo and exhaust temperature will indicate the exhaust stack may be blocked. There is no indication of high exhaust temperature given . This is plausible as it is one of the indications being monitored per the cautions of BwOP DG-11T2.

Task No: R-DG-015

Learn Objective: 4C.DG-06.

Reference: BwOP DG-11T2 rev 31

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

68

ID: RG10043-N01

Points: 1.00

Reactor power = 100%

- Core Burnup = 5000 EFPH
- Boron worth = -7.0 pcm/ppm
- CBD average Rod worth = 20 pcm/step
- Initial reactor coolant system (RCS) boron concentration = 600 ppm

The operating crew is planning a power descension to 50% power by boration/dilution.

The plan includes moving CBD from 220 to 195 steps inserted.

Which of the following is the final RCS boron concentration required to support the power ramp?
(Assume no change in core xenon reactivity.)

- A. 529 ppm
- B. 648 ppm
- C. 671 ppm
- D. 743 ppm

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 68

Reference Provided: BwCB-1 Figure 17A ALL pages rev24

History: Modified from GFE exam bank

RO Cog level: High

K/A- G 2.1.43 Importance RO 4.1 SRO 4.3

Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Tier: 3

Group: NA

K/A match: The question meets the KA because it requires the examinee to determine the effects of power defect, boron worth and, rod worth from the given information and use procedures (BwCB-1) to determine the final required reactor coolant boron concentration.

10CFR55.41(b)(1) Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Answer explanation and plausibility:

Choice A is incorrect. See choice C for the correct math. This is the answer you obtain if you don't do step 4 of the math below to account for the negative pcm change for adding boron, making this plausible, 529 ppm.

Choice B is incorrect. See choice C for the correct math. This is the answer you obtain if you use the wrong effective full power hour (EFPH) graph making this plausible, 648 ppm.

Choice C is correct.

- 1) Power is lowered 50% with a power defect of +1000pcm from page 3 of 4 of Fig 17A
- 2) Rods add 25 steps * -20 pcm/step= -500PCM
- 3) Power defect 1000 pcm plus rod worth -500 pcm = 500 pcm Net reactivity left
- 4) Net reactivity must be 0 pcm so 500 pcm + boron change =0 so boron change= -500pcm
- 5) -500 pcm /-7 pcm/ppm = 71.4 ppm boron change
- 6) 600 PPM boron + 71.4 PPM boron equals 671 ppm

Choice D is incorrect. See the math for C for the correct answer. This is the answer obtained if rods are omitted from the calculation, making this plausible, 743 ppm.

Task No: R-GP-050

Learn Objective: 4C.GP-01

Reference: BwCB-1 Fig 17A rev 24

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

69

ID: RG10023-N01

Points: 1.00

An operating surveillance is commenced on a Monday morning and will be kept open as an in-progress procedure for several days.

On Wednesday afternoon of the same week, an NSO assigned to continue work on the surveillance discovers that the procedure revision being used was superseded that Wednesday morning.

Per HU-AA-104-101, "PROCEDURE USE AND ADHERENCE," the in-progress surveillance...

- A. MUST be re-performed from the beginning with the latest revision. NO provision exists for using the superseded (old) revision.
- B. can ONLY continue if data and place keeping are transferred to the latest revision. NO provision exists for using the superseded (old) revision.
- C. can be completed using the superseded (old) revision. However, a temporary procedure change MUST be made to the old revision to match the current revision.
- D. can be completed using the superseded (old) revision. However, a supervisor MUST evaluate that the old revision can still be performed as written.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 69

Reference Provided: No

History: Bank question from 2011 NRC exam RO **Cog level:** Memory

K/A- G 2.2.14 **Importance** RO 3.9 SRO 4.3

G 2.2.14 Knowledge of the process for controlling equipment configuration or status.

Tier: 3

Group: NA

K/A match: The question requires the examinee to have knowledge of the fact that in progress procedures are used to maintain configuration and control of plant equipment. Combining that knowledge with knowledge of the process for procedure revisions issued during an in progress procedure meets the K/A for knowledge of controlling equipment configuration or status.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect. Section 3.1.3 allows for using the current revision with supervisor review and approval. This action would be the only action available if the supervisor found issue and didn't approve continuing with the previous revision, making this plausible.

Choice B is incorrect. Section 3.1.3 allows for using the current revision with supervisor review and approval. This would allow the operator to update the procedure revision working without repeating unaffected work, making this plausible.

Choice C is incorrect. Section 3.1.3 allows for using the current revision with supervisor review and approval. This is how a fatal flaw would be handled for a procedure needing immediate revision, making this plausible.

Choice D is correct. Section 3.1.3 of HU-AA-104-101 states that only current revisions of procedures can be used but then qualifies the statement to include superseded revisions for applicable work provided a supervisor review is performed to ensure no fatal flaws exist.

Task No: R-AM-133

Learn Objective: T.AM04-15

Reference: HU-AA-104-101 rev 5

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

70

ID: RG10056-N01

Points: 1.00

Unit 1 is in Mode 1.

- SAT 142-1 faults.

Unit 2 is in Mode 5.

- Both bus 241 and 242 are energized from offsite power.

(1) What Tech Spec LCO must Unit 1 enter?

(2) What Tech Spec LCO must Unit 2 enter, if any?

- A. (1) 3.8.9, Distribution Systems Operating
(2) None
- B. (1) 3.8.9, Distribution Systems Operating
(2) 3.8.2, AC Sources Shutdown
- C. (1) 3.8.1, AC Sources Operating
(2) 3.8.2, AC Sources Shutdown
- D. (1) 3.8.1 AC Sources Operating
(2) None

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 70

Reference Provided: No

History: Bank question from ILT bank RO Cog level: High

K/A- G 2.2.42 Importance RO 3.9 SRO 4.6

G 2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Tier: 3

Group: NA

K/A match: The question requires the examinee to determine the entry conditions for Tech Spec based on system parameters.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect. There is no indication that there is a fault on any of the unit 1 buses that would make them inoperable. Unit 2 has not lost any electrical equipment required by Tech Specs for the current mode. The fault on the SAT could be incorrectly viewed as being caused by a fault on an ESF bus making this plausible.

Choice B is incorrect. There is no indication that there is a fault on any of the unit 1 buses that would make them inoperable. Unit 2 has not lost any electrical equipment required by Tech Specs for the current mode. The fault on the SAT could be incorrectly viewed as being caused by a fault on an ESF bus and the loss of unit 1 offsite power would be entry criteria for Tech Spec if unit 2 was in mode 1-4, the requirements for mode 5,6 are less stringent, making this plausible.

Choice C is incorrect. Unit 2 has not lost any electrical equipment required by Tech Specs for the current mode. Unit 1 will be in Tech Spec LCO 3.8.1 for the loss of offsite power sources from unit 1 and the loss of unit 1 offsite power would be entry criteria for Tech Spec if unit 2 was in mode 1-4, the requirements for mode 5,6 are less stringent, making this plausible.

Choice D is correct. Unit 1 will be in Tech Spec LCO 3.8.1 for the loss of offsite power sources from unit 1 and Unit 2 has not lost any electrical equipment required by Tech Specs for the current mode.

Task No: R-AP-001

Learn Objective: 4C.AP-01

Reference: Tech Spec 3.8.1 & 3.8.2

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

71

ID: RG30004-N01

Points: 1.00

A General Emergency has been declared due to a Large Break LOCA.

- A High Radiation area needs to be accessed for LIFE SAVING activities.

Per RP-AA-203 the MAXIMUM TEDE dose the volunteer emergency responder can receive for this activity?

- A. 5 REM TEDE
- B. 10 REM TEDE
- C. 25 REM TEDE
- D. 50 REM TEDE

Answer: C

Answer Explanation

Question: 71

Reference Provided: No

History: Bank question 2009 NRC exam

RO

Cog level: Memory

K/A- G 2.3.4 **Importance** RO 3.2 SRO 3.7

G 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.

Tier: 3

Group: NA

K/A match: The question meets the K/A, requires examinee knowledge of radiation exposure limits under emergency conditions.

10CFR55.41(b)(12) Radiological safety principles and procedures.

Answer explanation and plausibility:

Choice A is incorrect. The exposure limit is 25 REM TEDE per RP-AA-203. This choice is the dose allowed annually for occupational worker limiting his total annual dose to 10 REM which is plausible since normal radiation exposure limits apply to the individual for the year.

Choice B is incorrect. This is the dose allowed for protecting valuable property per RP-AA-203 section 4.5.3. Exposure limit is "per activity". This is plausible because to receive this dose a person must volunteer and be granted permission the same as a person performing lifesaving activities.

Choice C is correct. The exposure limit is 25 REM TEDE per RP-AA-203. This is the dose for saving a life, limiting total annual dose to 25 REM.

Choice D is incorrect. The exposure limit is 25 REM TEDE per RP-AA-203. This is the allowed annual TODE dose for an occupational worker making this plausible.

Task No: R-AM-133

Learn Objective: 3E.AM-133

Reference: RP-AA-203 rev 5

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

72

ID: RG30011-C01

Points: 1.00

A Unit 1 Containment Release is in progress per RP-BR-980, CONTAINMENT VENT AND MINI PURGE GASEOUS EFFLUENTS.

RUNNING Aux Bldg Vent Stack exhaust fans:

- 0VA02CC, VA EXH FAN 0C TRN 0B
- 0VF01CA, FILTERED TANK VENTS EXH FAN 0A
- 0VW03CA, RADWASTE BLDG EXH FAN 0A
- 0VL02CA, LAB HVAC FUME HOOD EXH FAN 0A

ALL other Aux Bldg Vent Stacks exhaust fans and charcoal booster fans are shutdown.

Which of the following fan trips would require the operator to secure the Unit 1 Containment Release?

- A. 0VA02CC
- B. 0VF01CA
- C. 0VW03CA
- D. 0VL02CA

Answer: D

Answer Explanation

Question: 72

Reference Provided: No

History: Bank question on past cert exams

RO

Cog level: High

K/A- G 2.3.11 **Importance** RO 3.8 SRO 4.3
G 2.3.11 Ability to control radiation releases.

Tier: 3

Group: NA

K/A match: The question meets the K/A, requires examinee ability to control a containment release when conditions for the release are no longer met.

10CFR55.41(b)(13) Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Answer explanation and plausibility:

Choice A is incorrect, 0VA02CC discharges to U-2 stack, therefore would not affect U-1 release. The VA exhaust fans provide the largest flow through the vent stack making this a plausible distractor.

Choice B is incorrect, VF discharges to U-2 stack, therefore would not affect U-1 release. With no VA fan running in the unit 1 plenum, it is plausible that the unit 0 fans would all be credited for unit 1 vent stack flow.

Choice C is incorrect, 0VW03CA discharges to U-2 stack, therefore would not affect U-1 release. With no VA fan running in the unit 1 plenum, it is plausible that the unit 0 fans would all be credited for unit 1 vent stack flow.

Choice D is correct. VL fan trip would leave no fans discharging to the U-1 stack, which is not sufficient for a containment release, the release must be secured.

Task No: R-VQ-004

Learn Objective: 4C.VQ-01

Reference: RP-BR-980 rev 18 page 7

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

73

ID: RG32314-N01

Points: 1.00

An RCS LOCA resulted in core damage on Unit 1.

Which of the following evolutions have the LOWEST potential for raising radiation levels in the Auxiliary Building?

- A. Place one train of Unit 1 RH in service per BwOP RH-6, PLACING THE RH SYSTEM IN SHUTDOWN COOLING, without establishing RH letdown.
- B. Restart one RCP per 1BwOA ESP-1, REACTOR COOLANT PUMP STARTUP DURING ABNORMAL CONDITIONS, without unisolating seal return.
- C. Re-establish letdown per 1BwOA ESP-2, REESTABLISHING CV LETDOWN DURING ABNORMAL CONDITIONS.
- D. Establish cold leg recirculation per 1BwEP ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.

Answer: B

Answer Explanation

Question: 73

Reference Provided: No

History: Bank question 2013 NRC exam

RO

Cog level: Memory

K/A- G 2.3.14 Importance RO 3.4 SRO 3.8

2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Tier: 3

Group: NA

K/A match: The examinee must discern evolutions that will significantly raise radiation levels in areas that are NOT normally high radiation areas, during a core damaging event.

10CFR55.41(b)(12) Radiological safety principles and procedures.

Answer explanation and plausibility:

Choice A is incorrect. Placing RH in service without establishing RH letdown will still spread the radiation levels into the aux building RH piping. This evolution will be required to cooldown the unit to cold shutdown making this a plausible distractor.

Choice B is correct. Starting an RCP during abnormal condition per 1BwOA ESP-1 with seal return not realigned to the VCT will not bring any of the primary coolant out of the containment building to the aux building.

Choice C is incorrect. Establishing letdown will bring highly contaminated primary coolant into the aux building. Establishing letdown returns pressurizer level control and cleanup path to the demineralizers to cleanup the coolant, making this a plausible distractor.

Choice D is incorrect. Cold leg recirculation bring highly contaminated coolant from the containment floor into the Aux building to recirculate it through the entire ECCS system in the Aux building. This activity will be performed if the RWST is depleted to the low 2 RWST level and will not be stopped for high radiation level, making this a plausible distractor.

Task No: R-OA-121

Learn Objective: 4D.OA-82

Reference: I1-MI-XL-13 rev.3b

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

74

ID: RG40003-N01

Points: 1.00

What does a BLACK DOT in the upper Right-Hand corner of equipment identification labels on the MCR panels designate?

- A. Emergency Operating Procedure equipment
- B. Post-Accident Monitoring instrumentation
- C. FLEX Support Guideline equipment
- D. Remote Shutdown Panel instrumentation

Answer: B

Answer Explanation
<p>Question: 74 Reference Provided: No History: New question for the 2018 NRC exam RO Cog level: Memory K/A- G2.4.3 Importance RO 3.7 SRO 3.9 G 2.4.3 Ability to identify post-accident instrumentation Tier: 3 Group: NA K/A match: The question requires the examinee to identify how post-accident instrumentation is labeled. 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.</p> <p>Answer explanation and plausibility: Choice A is incorrect. Per 1BwOSR 3.3.3.1, the black dot is PAM related instrumentation. There are requirements to have equipment required for the EOPs specially labeled making this a plausible distractor. Choice B is correct. Per 1BwOSR 3.3.3.1, the black dot is PAM related instrumentation. Choice C is incorrect. Per 1BwOSR 3.3.3.1, the black dot is PAM related instrumentation. There are requirements to have equipment required for the FSGs specially labeled making this a plausible distractor. Choice D is incorrect. Per 1BwOSR 3.3.3.1, the black dot is PAM related instrumentation. The remote shutdown instrumentation Tech Spec 3.3.4 surveillance is frequently confused with the PAM surveillance, making this a plausible distractor.</p> <p>Task No: R-AM-014 Learn Objective: T.MI04-01 Reference: I1-MI-XL-04 MCD 1.3 Critical Parameters, 1BwOSR 3.3.3.1 rev 19</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

75

ID: RE1WE02-N01

Points: 1.00

Unit 1 is at 100% power.

- Pressurizer pressure channel 1PT-455 is in test.

Subsequently, pressurizer pressure transmitter 1PT-458 fails low causing a low pressurizer pressure SI.

Five minutes later, the crew has transitioned to 1BwEP ES-1.1, SI TERMINATION.

- The NSO depresses BOTH SI RESET pushbuttons.
- The BYPASS - PERMISSIVE LIGHTS silence, acknowledge, and reset pushbuttons are depressed.

Which of the following BYPASS - PERMISSIVE LIGHTS will be lit?

- A. SI ACTUATED
- B. PZR LOW PRESS SI BLOCK PERMISSIVE P11
- C. PZR SI BLOCKED TRN A
- D. AUTO SI BLOCKED

Answer: D

Answer Explanation

Question: 75

Reference Provided: No

History: Bank question 2009 NRC exam

RO

Cog level: High

K/A- G 2.4.46 **Importance** RO 4.2 SRO 4.2

G 2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

Tier: 3

Group: NA

K/A match: The question meets the K/A, requires examinee ability to interpret the significance of alarms by verifying alarms are consistent with plant conditions during SI termination.

10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Answer explanation and plausibility:

Choice A is incorrect, SI ACTUATED will go dark after reset pushbuttons are depressed and SI blocked will light. This prevents further automatic SI signals from actuating. Manual SI will still re-actuate SI even with SI blocked. The condition that caused the initial SI is still present, making this a plausible distractor.

Choice B is incorrect, channel 458 does not input into the 2/3 logic for P11. P11 will light when 2 of 3 pressurizer pressure channels are below 1930 psig, making this a plausible distractor.

Choice C is incorrect, this alarm lights when pressure is <P11 and the SI lo press SI block switch is taken to block. This is plausible since the PZR SI signal is auto blocked from actuating SI once the SI reset buttons are pressed.

Choice D is correct, resetting SI later than 1 minute after an auto SI will block auto SI and light the AUTO SI BLOCKED alarm.

Task No: R-EP-036

Learn Objective: 3D.EP-02-E

Reference: BIG NOTE EF-4 rev 5

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

76

ID: SS20004-N02

Points: 1.00

Unit 1 is at 90% power.

LTDWN HX OUT TEMP CONTROL 1CC130A, 1TK-0130, causes letdown temperature to lower to 75°F.

Assuming NO operator actions are taken, the VCT boron concentration will (1), AND the correct procedure the US will use to mitigate the event is (2).

- A. (1) lower
(2) 1BwOA PRI-12, UNCONTROLLED DILUTION.
- B. (1) rise
(2) 1BwOA PRI-6, COMPONENT COOLING MALFUNCTION.
- C. (1) lower
(2) 1BwOA PRI-6, COMPONENT COOLING MALFUNCTION.
- D. (1) rise
(2) 1BwOA PRI-12, UNCONTROLLED DILUTION.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question 76

Reference provided: NO

History: Modified from 2009 NRC #SS20004-N01 **SRO Cog Level:** High

K/A 004 A2.10 RO 3.9 SRO 4.2

004 Chemical and Volume Control System

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Uncontrolled boration or dilution

TIER: 2

GROUP: 1

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

K/A match: The question meets the K/A, it requires examinee ability to predict impact of malfunction on CVCS (letdown temperature control) and select the correct procedure to use to mitigate the consequences.

SRO Justification: The question is SRO level because it requires assessment of conditions and selection of appropriate procedure. The conditions are not entry criteria or immediate actions of either procedure and not part of the overall mitigative strategy.

Answer explanation and Plausibility:

Choice A is correct, 1BwOA PRI-12, step 3, will check for low letdown temperature and then manually control letdown temperature or bypass the demins by placing 1CV129 to the VCT position. This will stop the dilution.

Choice B is incorrect. This is plausible because letdown temperature effects on boron concentration are a common misunderstanding. The affected controller is a CC flow controller on the CVCS system and could plausibly be fixed utilizing 1BwOA PRI-6. The hi letdown temperature alarm is checked in 1BwOA PRI-6 step 6 to ensure proper temperature control.

Choice C is incorrect. This is plausible because the boron concentration will drop and the affected controller is a CC flow controller on the CVCS system and could plausibly be fixed utilizing 1BwOA PRI-6. The hi letdown temperature alarm is checked in 1BwOA PRI-6 step 6 to ensure proper temperature control.

Choice D is incorrect, This is plausible because letdown temperature effects on boron concentration are a common misunderstanding. The letdown temperature is checked greater than 80F in 1BwOA PRI-12 step 3 to ensure proper temperature control and will correct the temperature issue.

Task No: S-OA-065

Obj No: 7D.OA-021-A

reference: 1BwOA PRI-12 rev 101

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

77

ID: SS20005-N01

Points: 1.00

Unit 1 is in MODE 4, 275°F RCS cold leg temperature, 300 psig.

- RCS boron concentration is 1980 ppm.
- Plant heatup is in progress following a refuel outage.
- ALL RCPs are available.
- ONLY 1D RCP is running.
- 1A RH train is running in shutdown cooling.
- 1B RH train is aligned for cold leg injection.
- 1D SG NR level is 25%.
- 1A, B & C SG NR levels are all approx. 50%.

Preparations to start 1C RCP per BwOP RC-1, STARTUP OF A RCP, are in progress.

The 1A RH train is being secured from S/D cooling and aligned for injection per BwOP RH-11, SECURING THE RH SYSTEM FROM SHUTDOWN COOLING.

Prior to entering mode 3, to ensure the 1A RH train is operable per Technical Specification 3.5.2...

- A. one additional RCP MUST be in operation.
- B. 1A RH train temperature MUST be lowered.
- C. 1D SG NR level MUST be raised.
- D. 1A RH train boron concentration MUST be raised.

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 77

Reference Provided: No

History: Bank question from 2013 cert # RS10005-C01 **SRO Cog level: High**

K/A-005 2.1.23 RO 4.3 SRO 4.4

SYS 005 Residual Heat Removal

G 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Tier 2

Group 1

K/A match: Question meets KA - question requires examinee knowledge of procedural requirement for RH adequate subcooling prior to placing RH in injection alignment.

SRO Justification: SRO level question requires assessment of plant conditions (normal, abnormal or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed.

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because T.S 3.4.6 "RCS Loops- MODE 4" requires two loops of RC or RH operable but only one in operation. The 1D RCP running satisfies the loop in operation and the standby RCS loops satisfy the second operable loops.

Choice B is correct. Per BwOP RH-11, the RH precautions and limitations, suction temperature must be less than 260°F before the RH suction valve from the RWST can be opened to prevent the pump from becoming inoperable due to steam binding. For recirculation capability the temperature limit is 200°F in order to call the RH train available for TS 3.5.2

Choice C is incorrect, LCO 3.4.6 only requires $\geq 18\%$ NR level for the RCS loop in operation in MODE 4. This is plausible since SG NR level is normally maintained at 60% +/- 5% at power.

Choice D is incorrect, there is no requirement for the 1A RH train to be borated prior to aligning for injection. MODE 4 only requires one ECCS train which is satisfied by the 1B RH train. This is plausible since there is a requirement to ensure the RH train is borated to the COLR limit in the alignment for shutdown cooling procedure BwOP RH-6. However, that is not a requirement for securing from shutdown cooling.

Task No: R-RH-004

Learn Objective: S.RH1-09-A

Reference: T.S. bases 3.5.2

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

78

ID: SS10022-N01

Points: 1.00

Unit 1 at 100% power.

- Lake temperature is 90°F.
- CNMT CHILLER 1A is running.
- 1WO01PA, CHILLED WATER Pump 1A, has tripped due to an internal ground.

The CNMT CHILLER prevents challenging the containment design limit for _____ (1) _____ in the event of a MSL break.

AND

To prevent challenging this design limit due to the chilled water pump trip, the US will _____ (2) _____.

- A. (1) temperature
(2) order a containment release package
- B. (1) pressure
(2) direct the NSO to swap VP trains
- C. (1) temperature
(2) direct the NSO to swap VP trains
- D. (1) pressure
(2) order a containment release package

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 78

Reference Provided: No

History: Modified from 2013 NRC RS10022-N01

SRO Cog level: Memory

K/A-022A2.06 RO 2.8 SRO 3.2

SYS 022 Containment Cooling System

A2.06 Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of CCS

Tier 2

Group 1

K/A match: Examinee must predict the impact on the containment accident analysis and based on that prediction determine the correct procedurally required action.

SRO Justification: SRO level question requires assessment of plant conditions (normal, abnormal or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. The impact must be determined from knowledge of the basis for containment temperature limits in Tech Specs.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect, a containment package will allow a release or feed and bleed to be started but will not effectively remove the heat from containment. TS 3.6.5 containment temperature basis describes the limiting case for containment temperature and pressure being related. The basis for containment temperature is a main steam line break making this plausible.

Choice B is incorrect, TS 3.6.5 containment pressure limit is based on a primary LOCA, containment temperature basis describes the limiting case for containment temperature and pressure being related. The basis for containment temperature is a main steam line break. This is plausible because the correct answer is to swap VP chiller trains and pressure will rise as a result of temperature rise.

Choice C is correct, TS 3.6.5 containment temperature basis describes the limiting case for containment temperature and pressure being related. The basis for containment temperature is a main steam line break while containment pressure limit is a primary LOCA. The 1B chilled water pump cannot supply the 1A train chiller, although the piping is cross-connected, it is not cross-connected in a geometry that facilitate cross-train operation (see Big Note WO-2). The crew must therefore swap to the 1B chiller system. This must be done to prevent containment temperature from rising to the 120F TS limit which is based on a MSLB.

Choice D is incorrect, TS 3.6.5 containment pressure limit is based on a primary LOCA, containment temperature basis describes the limiting case for containment temperature and pressure being related. The basis for containment temperature is a main steam line break. This is plausible because temperature and pressure are related containment pressure will go up when the chilled water pump trips and a containment release will keep containment pressure lower but will not keep temperature low.

Learn Objective: S.W01-02-B.

Reference: T.S. bases 3.6.5, BwAR 0-33-D3, CNMT CHLR UNIT TROUBLE rev 7. BwOP VP-1, RCFC REFRIGERATION UNIT AND CHILLED WATER SYSTEM STARTUP rev. 55. Big Note WO-2 rev. 5.

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

79

ID: SS20062-N02

Points: 1.00

Both units are 100% power.

Breaker 2414 is OOS for maintenance with cubicle damage.

Unit 1 experiences a loss of all AC power.

- Bus 142 is faulted.
- Breaker 1413 failed to close automatically AND manually.
- Unit 2 has NOT been affected.

1BwCA-0.0, LOSS OF ALL AC POWER, is in progress at step 9, CROSSTIE BUS 142 TO UNIT 2.

What procedure(s) will the US use to respond?

1BwCA-0.1, LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED.

1BwCA-0.0, ATTACHMENT B, EXTENDED LOSS OF ALL AC POWER RESPONSE.

0BwFSG-5, INITIAL ASSESSMENT AND FLEX EQUIPMENT STAGING.

- A. ONLY 1BwCA-0.1.
- B. ONLY 0BwFSG-5.
- C. ONLY 1BwCA-0.0, ATTACHMENT B.
- D. 1BwCA-0.0, ATTACHMENT B, and 0BwFSG-5, performed concurrently.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 79

Reference Provided: No

History: New question for 2018 NRC exam

SRO Cog level: High

K/A-062A2.06 RO 3.4 SRO 3.9

SYS 062 AC Electrical Distribution

A2.06 Ability to (a) predict the impacts of the following malfunctions or operations on the AC Electrical Distribution; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Keeping the safeguards buses electrically separated.

Tier 2

Group 1

K/A match: The question requires predicting the impact of not being able to crosstie safeguards buses between unit 1 and unit 2. Using this prediction, the SRO will determine the appropriate procedure and actions to mitigate the consequences.

SRO Justification: SRO level question requires assessment of plant conditions (normal, abnormal or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect, because bus 141 crosstie breaker from unit 2 is out of service and the DG breaker 1413 cannot be closed. This is plausible because no conditions are present that would warrant SI actuation and because unit 2 has both safeguards buses energized to provide unit 1 power.

Choice B is incorrect. This is plausible because 1BwCA-0.0 ATTACHMENT B will be used to recover however; it is used in conjunction with 0BwFSG-5.

Choice C is incorrect. This is plausible because 0BwFSG-5 will be used to recover however; it is used in conjunction with 1BwCA-0.0 ATTACHMENT B.

Choice D is correct, at step 9 of 1BwCA-0.0 with bus 142 faulted the RNO of step 9.b.3. directs the operator to go to 0BwFSG-5 while continuing with 1BwCA-0.0 ATTACHMENT B. The 0BwFSG-5 will begin to setup FLEX strategy power supplies while 1BwCA 0.0 ATTACHMENT B takes actions to maximize available inventory and prepare for FLEX power supplies.

Task: S-CA-005

Learn Objective: S-CA-005.

Reference: 1BwCA 0.0 rev301

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

80

ID: SS20063-N01

Points: 1.00

Unit 1 is at 100% power.

DC bus 111 de-energizes.

- The crew manually trips the reactor, enters 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, and performs the immediate actions.
- Concurrently, the US has entered 1BwOA ELEC-1, LOSS OF DC BUS.

What is the FIRST local action the US will direct an EO to perform?

- A. TRIP a TDFW pump.
- B. CLOSE the de-energized 4 kv bus SAT feed breaker.
- C. TRIP the PMG output breaker.
- D. CHECK status of 125V DC Feed from Battery 111 (breaker AF2).

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 80

Reference Provided: No

History: New Question SRO Cog level: Memory

K/A063G2.4.49-Importance RO 4.6 SRO 4.4

0063 DC Electrical Distribution

Generic 2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Tier: 2

Group: 1

K/A match: This question requires the operator to determine, without references, those actions required to be performed manually due to a loss of DC electrical distribution control to prevent further damage to plant systems or components.

SRO Justification: The question requires knowledge beyond immediate actions, system knowledge, entry criteria and mitigative strategy. This requires knowledge of diagnostic steps and decision point in the EOPs that involve transitions to event-specific sub-procedures.

(1BWOA ELEC-1)

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because the TDFW pump will attempt to draw steam from the main steam lines and cause a cooldown in the primary and because this is done following tripping the PMG output breaker.

Choice B is incorrect. Both ESF busses will be energized from the SAT, both before and after the manual trip and not require actions to be performed by an EO in the field. This is plausible because ensuring 4KV ESF buses are energized is part of the immediate actions of 1BwEP-0, and because the loss of DC bus 111 removes control power to bus 141.

Choice C is correct. 1BWOA ELEC-1 ATTACHMENT A step 2 trips the PMG output breaker locally. This ensure that damage is not done to the main generator or any of the other equipment tied to it as it begins motoring post trip.

Choice D is incorrect. Per 1BWOA ELEC-1, this will not be done until after the PMG output breaker and the TDFW pumps are tripped. Investigating the cause of the loss will be a high priority for restoration of DC power and better control of the plant, making this a plausible distractor.

Task No: S-OA-002

Learn Objective: 8D.OA-001-B

Reference: 1BWOA ELEC-1 rev111

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

81

ID: SS20029-N01

Points: 1.00

Outage and maintenance activities are underway in containment that could result in a rising trend in airborne radioactivity.

- Containment integrity is being maintained.
- Current containment pressure is + 0.5 psig.

If an extended feed and bleed is required per BwOP VQ-6, CONTAINMENT MINI-PURGE SYSTEM OPERATION, the US will direct the NSO to delay starting the 1VQ04C, MINI-PURGE SUPPLY FAN,...

- A. until containment integrity is NO longer required.
- B. because containment pressure is currently too LOW.
- C. because containment pressure is currently too HIGH.
- D. until the mini-purge exhaust fan is running.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 81

Reference Provided: No

History: New question for 2018 NRC exam

SRO Cog level: Memory

K/A 029A2.01 - Importance RO 2.9 SRO 3.6

029 Containment Purge System A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the Containment Purge System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Maintenance or other activity taking place inside containment.

Tier: 2

Group: 2

K/A match: This matches the K/A because it requires the examinee to consider the impact of maintenance activities on containment radiation levels and predict what impact that activity will have on the use of the containment purge system.

SRO Justification: This question does not rely on system, immediate action, entry condition or general mitigative knowledge to answer the question. The question requires the selection of the correct procedure section to mitigate the containment pressure before beginning to perform the desired actions.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect. The fact that the containment purge supply and exhaust system has automatic containment isolation valves make it possible to open these paths while containment integrity is set and that makes this choice incorrect. This is plausible because it is similar to opening both containment airlocks during an outage, which is only allowed when containment integrity is not required; opening the purge system supply and exhaust dampers also opens containment to atmosphere.

Choice B is incorrect. The pressure is higher than the required pressure to start the forced air supply fan per the precaution, making this incorrect. This is plausible because for air to escape from containment a differential pressure with the outside atmosphere is required. The higher the pressure in containment the higher the air flow will be out of containment. It is plausible that a higher pressure would provide better air turnover.

Choice C is correct. In order to address the rising activity caused by the maintenance activities, it is necessary to run the mini-purge supply fan. BwOP VQ-6 limitation E.9 and notes prior to starting mini-purge supply fan direct the operator to ensure containment pressure is < 0.3 psig to ensure forward flow. Past OPEX has shown running fans without forward flow has damaged fan blading. BwOP VQ-6 contains independent steps for operating the mini-purge system during extended purge ("feed and bleed") operations. This allows the operator to lower pressure by operating the exhaust part of the system first to lower pressure.

Choice D is incorrect. Other ventilation systems have interlocks that require exhaust fans to be running before supply fans. It is plausible that since containment pressure rises when integrity is set that this system would have the same interlock. This is not the case.

Task No: S-HP-002

Learn Objective: 7C.HP-002-A

Reference: BwOP VQ-6 rev 25

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

82

ID: SS20033-N01

Points: 1.00

Both Units are 100% power.

FC pumps were swapped from Unit 1 to Unit 2.

- The newly started FC cooling pump develops a leak between the pump and discharge check valve.

Spent Fuel Pool level drops and stabilizes as indicated below.



The Shift Manager will declare a/an...

- A. Unusual Event.
- B. Alert.
- C. Site Area Emergency.
- D. General Emergency.

Answer: B

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 82

Reference Provided: Yes EAL matrix EP-AA1001 Addendum 3 rev 3 pages 2-1 through 2-24

History: New for 2018 NRC Exam SRO **Cog level:** High

K/A-033G2.4.41

033 Spent Fuel Pool Cooling

G 2.4.41 Knowledge of the emergency action level thresholds and classifications. Importance RO 2.9 SRO 4.6

Tier: 2

Group: 2

K/A match: The question asks for examinee to interpret information on spent fuel pooling cooling accident and determine the EAL threshold.

SRO Justification: Interpreting EAL thresholds is an SRO job function.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because an Unusual Event is a potential EAL declaration for lowering SFP level and because RU2 is declared for an unplanned drop in SFP water level as indicated in the question stem. This is incorrect because the SFP level indicating 9' would require declaring an alert.

Choice B is correct. Spent fuel pool level less than 10' is the threshold value for RA2.

Choice C is incorrect. This is plausible because a Site Area Emergency is a potential EAL declaration for lowering SFP level would be declared per RS2 if level had continued to lower less than 1'. Because indicated SFP is less than 10', an error reading the meter or reading and applying the level limit of EP-AA-1001 Addendum 3 is possible.

Choice D is incorrect. RG2 is plausible because a General Emergency is a potential EAL declaration for lowering SFP level and the Emergency Director would declare the General Emergency promptly if SFP level were below 1' and it was determined that level would not be restored promptly. Furthermore, because the indicated SFP is less than 10', an error reading the meter or reading and applying the level limit of EP-AA-1001 Addendum 3 is a possibility.

Task No:

Learn Objective: S.FC1-05

Reference: EP-AA-1001 Addendum 3 rev 3

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

83

ID: SS20056-N01

Points: 1.00

Unit 1 is at 100% power.

- A main condenser tube leak is in progress and troubleshooting efforts are ongoing.
- Chemistry reports secondary conductivity has exceeded ACTION LEVEL 2 limits.

Per 1BwOA SEC-2, ABNORMAL SECONDARY CHEMISTRY, the MAXIMUM allowed power is...

- A. 100%.
- B. < 50%.
- C. < 5%.
- D. 0% (MODE 3).

Answer: B

Answer Explanation

Question: 83

Reference Provided: No

History: New question for 2018 NRC exam

SRO Cog level: Memory

K/A-056A2.05 Importance RO 2.1 SRO 2.5

056 Condensate A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operation: Condenser tube leakage.

Tier: 2

Group: 2

K/A match: This question requires the interpretation of the indications of a condenser tube leak and the determination of procedurally required actions to mitigate those indications.

SRO Justification: SRO level question requires assessment of plant conditions (normal, abnormal or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because chemistry action level 1 allows continued operations for a limited time period. Action level 2 does not allow this continued operation.

Choice B is correct. Per step 10a of 1BwOA SEC-2, power must be reduced to <50% within 24 hours.

Choice C is incorrect. This is plausible as the first power level checked in 1BwOA SEC-2 is 5% and power will be held at this level in any action level. This is incorrect since the unit was already at 100% power.

Choice D is incorrect. This is plausible if the unit remains in action level 2 for greater than 300 hours and a level 3 threshold is present, then the unit will need to be shut down to mode 3 in 6 hours. Since the action level was just reported this is not the requirement.

Task No: S-OA-097

Learn Objective: 7D.OA-036-A

Reference: 1BwOA SEC-2 rev 103

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

84

ID: SE10011-N02

Points: 1.00

Unit 1 experienced a Large Break LOCA.

- 1BwEP ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, is in progress at step 2, CHECK CNMT SUMP LEVEL.
- The STA reports a RED path in the Heat Sink status tree exists.

Which of the following procedure flow paths will the US follow?

- A. Complete step 6, START ECCS PUMPS AS NECESSARY, of 1BwEP ES-1.3, and then implement 1BwFR H.1, LOSS OF SECONDARY HEAT SINK.
- B. Immediately go to 1BwFR H.1, LOSS OF SECONDARY HEAT SINK, then return to 1BwEP ES-1.3 when complete.
- C. Complete ALL steps of 1BwEP ES-1.3 and then implement 1BwFR H.1, LOSS OF SECONDARY HEAT SINK.
- D. Immediately go to 1BwFR H.1, LOSS OF SECONDARY HEAT SINK, while concurrently aligning ECCS for Cold Leg Recirculation per 1BwEP ES-1.3.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 84

Reference Provided: No

History: Bank question BWLI-EP1-113 **SRO** **Cog level:** Memory

K/A- EPE11 G2.4.20 Importance RO 3.8 SRO 4.3

EPE11 Large Break LOCA G2.4.20 Knowledge of operational implications of EOP warnings, cautions, and notes.

Tier: 1

Group: 1

K/A match: This question test the SRO knowledge of notes in 1BwEP ES-1.3 regarding functional restoration procedure use.

SRO Justification: This question does not use system, immediate action, entry condition or overall strategy knowledge to determine transition and decision points in the EOPS specific procedure transitions and use of EOP sub-procedures.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is correct. Per the 3rd note before step 1 of 1BwEP ES-1.3 BwFRs should NOT be implemented prior to completion of step 6.

Choice B is incorrect. This is plausible, per BwAP 340-1 step C.4.d.4 " In all other cases, a Red condition requires implementation of the designated _BwFR immediately upon detection. Any other procedure remains in suspension until the _BwFR is complete." 1BwEP ES-1.3 is one of the exceptions per earlier wording in the same section of BwAP 340-1.

Choice C is incorrect. This is plausible because the importance of completing the alignment for recirculation prior to performing other action is highlighted in BwAP "_BwES 1.3 has the operators establish Cold leg Recirculation in order to supply water to the ECCS pumps and return to or implement the BwFRs". This is incorrect to finish the procedure though per the note only steps 1 through 6 are required to be complete before implementing

Choice D is incorrect. This is plausible, red path procedures take priority per BwAP 340-1 and cold leg recirc is a high priority exception to performing functional restoration. Per the last note before step 1 of 1BwEP ES-1.3, BwFRs should not be implemented until step 6 is complete therefore this answer is incorrect.

Task No: S-EP-045

Learn Objective: T.EP02-01-C

Reference: 1BwEP ES-1.3 rev 300; BwAP 340-1 rev 29

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

85

ID: SS10010-N01

Points: 1.00

Unit 1 is at 20% power.

- PZR pressure is 2205 psig and lowering.
- 1RY455C, PZR Spray Valve, is open and NOT responding in manual NOR in auto control.
- All other PZR system components are operating as designed.

The Unit Supervisor will direct the crew to...

- A. manually trip the reactor, enter 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, and STOP the **1C** RCP.
- B. manually trip the reactor, enter 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, and STOP the **1D** RCP.
- C. STOP the **1C** RCP per BwOP RC-2, SHUTDOWN OF A RCP, and verify all PZR heaters energized.
- D. STOP the **1D** RCP per BwOP RC-2, SHUTDOWN OF A RCP, and verify all PZR heaters energized.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 85

Reference Provided: No

History: Bank (Braidwood NRC Exam 2011 Question # 87) **SRO** **Cog level: High**

K/A-APE27AA2.06 Importance RO 3.5 SRO 3.9

APE27 Pressurizer Pressure Control System (PZR PCS) Malfunction AA2.06 Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: Conditions requiring plant shutdown

Tier: 1

Group: 1

K/A match: The question requires the examinee to evaluate a failure of the pressurizer pressure control system that requires the unit to be shutdown to mitigate the issue.

SRO Justification: SRO level question requires assessment of plant conditions (normal, abnormal or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is correct. 1RY455C spray valve failing open requires that the RX be tripped in order to secure the 1C RCP to stop the forward flow through that valve. This is the procedurally directed action from 1BwPR 1-12-RY.

Choice B is incorrect. This is plausible because it follows the procedurally required actions. 1RY455B is the valve associated with the 1D RCP, the mismatched nomenclature and reverse order of the valve numbering make it plausible the examinee would choose this distractor.

Choice C is incorrect. This is plausible because the reactor trip on low flow in 1 of 4 loops is disabled below 30% power. An RCP could be secured without generating a trip and securing 1C RCP would stop flow through the 1RY455C.

Choice D is incorrect. This is plausible because the reactor trip on low flow in 1 of 4 loops is disabled below 30% power. An RCP could be secured without generating a trip and 1RY455B is the valve associated with the 1D RCP, the mismatched nomenclature and reverse order of the valve numbering make it plausible the examinee would choose this.

Task No: S.OA-028

Learn Objective: T.OA11-21

Reference: 1BwPR 1-12-RY rev 1 Draft

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

86

ID: SE10029-N01

Points: 1.00

The crew is responding to a RED PATH on the Core Cooling status tree.

- 1BwFR-C.1, RESPONSE TO INADEQUATE CORE COOLING, is in progress.
- The STA identifies a RED PATH on the Subcriticality status tree.

The US will 1 , in order to prevent 2 .

- A. (1) transition to 1BwFR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS
(2) severe RCS over PRESSURIZATION condition
- B. (1) transition to 1BwFR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS
(2) severe RCS OVERTEMPERATURE condition
- C. (1) continue in 1BwFR-C.1, RESPONSE TO INADEQUATE CORE COOLING
(2) severe RCS over PRESSURIZATION condition
- D. (1) continue in 1BwFR-C.1, RESPONSE TO INADEQUATE CORE COOLING
(2) severe RCS OVERTEMPERATURE condition

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 86

Reference Provided: No

History: New for 2018 NRC exam

SRO Cog level: Memory

K/A EPE029 G2.4.22 -Importance RO 3.6 SRO 4.4

EPE029 Anticipated Transient without scram G2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Tier: 1

Group: 1

K/A match: The question requires the examinee to determine the priority of functional restoration procedures and their basis.

SRO Justification: This question is SRO because it requires knowledge of administrative procedures that specify hierarchy, implementation, and/ or coordination of plant normal, abnormal, and emergency procedures.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is correct. Per BwAP 340-1 Nuclear power generation is a higher priority than core cooling. This is because the plant is not designed to remove more heat than residual heat post trip. It is assumed in all the other safety functions that the reactor is shutdown and not producing heat. The heat generates a high pressure that causes a loss of ECCS flow.

Choice B is incorrect. This is plausible because 1BwFR-S.1 is the higher priority procedure and the reactor not being shutdown generates higher temperatures. It is the high pressure generated by the temperature rise that is the basis for this being the higher priority.

Choice C is incorrect. This is plausible because 1BwFR-C.1 has been implemented and is taking actions to lower core exit temperature. The high core exit temperatures generate a high pressure that causes a loss of ECCS flow. The assumption in taking actions per 1BwFR-C.1 is that those actions are only compensating for decay heat, not power generation, making this answer incorrect.

Choice D is incorrect. This is plausible because 1BwFR-C.1 has been implemented and is taking actions to lower core exit temperature. With degraded core cooling, temperatures are approaching the point of damage to the core and peak clad temperatures can be seen during this type of accident. The assumption in taking actions per 1BwFR-C.1 is that those actions are only compensating for decay heat not power generation, making this answer incorrect.

Task No: S-AM-128

Learn Objective: 7E.AM-128-A

Reference: BwAP 340-1 rev 29 1BwFR-C.1 rev 300

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

87

ID: SE10057-N02

Points: 1.00

Unit 1 is at 100% power.

Inverter 112 failed.

- 1BwOA ELEC-2, LOSS OF INSTRUMENT BUS, step 2, LOCALLY ENERGIZE AFFECTED INSTRUMENT BUS, is in progress following a loss of Instrument Bus 112.
- Instrument Bus 112 has been RE-ENERGIZED per step 2.

The next action the US will direct is to...

- A. GO TO 1BwOA ELEC-2 ATTACHMENT B, INSTRUMENT BUS 112 ACTIONS.
- B. perform 1BwOA INST-2, OPERATION WITH A FAILED INSTRUMENT CHANNEL, for the affected channels.
- C. dispatch an operator to fail open Train B aux feed flow control valves by failing air to the valve operators.
- D. SHUTDOWN INVERTER 112 per BwOP IP-2, TRANSFERRING AN INSTRUMENT BUS FROM THE INVERTER TO THE CONSTANT VOLTAGE TRANSFORMER.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 87

Reference Provided: No

History: New question for 2018 NRC Exam

SRO Cog level: High

K/A APE057AA2.14 -Importance RO 3.2 SRO 3.6

APE057 Loss of Vital AC Electrical Instrument Bus, AA2.14 Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: That substitute power sources have come on line on a loss of initial AC.

Tier: 1

Group: 1

K/A match: The question requires the examinee to determine that substitute power sources have come online and which steps are required to be performed if the Instrument Bus is restored.

SRO Justification: SRO level question requires assessment of plant conditions (normal, abnormal or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because actions for losing instrument bus 112 are contained in this procedure. Many procedures have actions for recovery included in their attachments. The step for powering the instrument bus directs shutting down the inverter, and the crew will not perform the attachment if the instrument bus is restored making this choice incorrect.

Choice B is incorrect. This is plausible if a instrument or control failed as a result loss of the instrument bus, checking controlling channels is the first step on the loss of an instrument bus. The controlling instrumentation has more than one power supply and none are reported as failing making this incorrect.

Choice C is incorrect. This is plausible if instrument busses were setup like DC buses. Instrument bus 114 would be powered from 112, these are often confused as they share similar nomenclature (IB 112/DC 112). Instrument bus 114 provides power to the train B ESF actuation circuitry and would require the train B AF005 valves to be failed open as they would close on AF actuation. Because the busses are not tied together like DC buses, this is an incorrect answer.

Choice D is correct. Once power is restored from the alternate source, the inverter is secured to prevent further damage or loss of the bus. No further actions are required as all instrument powered items reenergize in their required state.

Task No: S-OA-004

Learn Objective: T.OA02-03

Reference: 1BWOA ELEC-2 rev 109 draft

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

88

ID: SEWE11-N-01

Points: 1.00

An RCS LOCA occurred from full power on Unit 1.

- 1B RH pump is OOS.

During performance of 1BwEP ES-1.3, COLD LEG RECIRCULATION.

- 1SI8811A, CNMT SUMP 1A ISOL VLV, position lights were dark.
- An EO was dispatched to investigate.

1BwCA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, step 7, DETERMINE CS REQUIREMENTS, is in progress.

After the EO completes field actions, the conditions pictured below are present.



The US will...

- TRANSITION to 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.
- RETURN to 1BwEP ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.
- CONTINUE in 1BwCA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.
- TRANSITION to 1BwCA-1.3, SUMP BLOCKAGE CONTROL ROOM GUIDELINE.

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer: B

Answer Explanation
<p>Question: 88 Reference Provided: No History: New for the 2018 NRC exam SRO Cog level: High K/A WE11 G2.1.31 -Importance RO 4.6 SRO 4.3 WE11 Loss of Emergency Coolant Recirculation G2.1.31 Ability to locate the relevant control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup Tier: 1 Group: 1 K/A match: The question requires the examinee to determine the status of the controls for recirculation to determine the flow path through the contingency action for loss of emergency coolant recirculation. SRO Justification: The examinee needs knowledge of diagnostic steps and decision points in the emergency operating procedures (1BwCA-1.1) that involve transitions to event-specific sub-procedures or emergency contingency procedures to restore emergency coolant recirculation 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Answer explanation and plausibility: Choice A is incorrect. 1BwEP-1 was the procedure in effect when transition to 1BwEP ES-1.3 would have been made. This is a plausible return point to continue the lineup of the plant and selection of the appropriate procedure for long term recovery. Without completing 1BwEP ES-1.3 ECCS flow will not be available for core cooling making this an incorrect choice. Choice B is correct. Per 1BwCA-1.1 with 1CV8804A open and 1SI8811A energized you will return to the procedure and step in effect. Choice C is incorrect. With 1SI8811A indicating open, it is plausible that the examinee would continue in 1BwCA-1.1 to continue efforts to ensure a suction source available. 1BwCA-1.1 does not require 1SI8811A to be open merely to be energized and available. The valve was closed without power, the open indication shows the valve is no longer closed. Choice D is incorrect. The closure of 1SI8811A would cause issues similar to sump blockage. This would be a plausible choice to attempt to restore core cooling. However, with RH suction still aligned to the RWST, actions in 1BwCA-1.1 will preserve RWST inventory and continue to provide core cooling making this choice incorrect.</p> <p>Task No: S-CA-012 Learn Objective: 7D.CA-002-A Reference: 1BwCA 1.1 rev 300</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

89

ID: SE20035-N01

Points: 1.00

Unit 1 experienced an RCS LOCA from 100% power.

1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, is in progress at step 15, VERIFY TOTAL AF FLOW.

- Containment pressure is 12 psig and rising.
- CETCs indicate 720°F and rising.
- RCS pressure is 190 psig and stable.
- S/G pressures are 1075 psig and stable.
- Both AF pumps failed to start and CANNOT be manually started.
- Both CV pumps and both SI pumps failed to start.

S/G levels (NR):

- 1A S/G 25%
- 1B S/G 24%
- 1C S/G 26%
- 1D S/G 30%

The procedure flowpath the US will use to address the above conditions is...

- A. 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.
- B. 1BwFR-C.2, RESPONSE TO DEGRADED CORE COOLING, ONLY.
- C. 1BwFR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, ONLY.
- D. 1BwFR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, then after step 1, 1BwFR-C.2, RESPONSE TO DEGRADED CORE COOLING.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 89

Reference Provided: No

History: 2016 BWD NRC exam question 93

SRO Cog level: High

K/A W/E05EA2.05 Importance RO3.4 SRO 4.4

W/E05 Loss of Secondary Heat Sink A2.05 Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Facility conditions and selection of appropriate procedure during abnormal and emergency procedures.

Tier: 1

Group: 1

K/A match: The question requires the examinee to interpret facility conditions to determine that a loss of heat sink exists and that the loss of heat sink is not the highest priority abnormal condition.

SRO Justification: This question requires knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect. Indication of a LOCA are present and a transition to 1BwEP-1 will be made within the EP procedures which makes this a plausible distractor. The next procedure entered will be in the FR procedures to 1BwFR-H.1 followed by 1BwFR-C.2 making this answer incorrect.

Choice B is incorrect. The highest safety function that is not being met is the core cooling. CETC greater than 700F is an orange path on the core cooling status tree. This makes this a plausible choice; however, the red path in Heat Sink is a higher priority than the orange path and status trees are not implemented until the first transition out of 1BwEP-0, so this answer is incorrect.

Choice C is incorrect. With containment at 12 psig, adverse containment values are to be used, with no AF flow and all SG NR levels <31%, 1BwEP-0 step 15 RNO requires going to 1BwFR-H.1 to restore heat sink, making this plausible. Step 1 of 1BwFR-H.1 checks if RCS pressure is above the SG pressure, this is to verify that the SG will act as a heat sink. Which is not true making this answer incorrect.

Choice D is correct. With containment at 12 psig, adverse containment values are to be used, with no AF flow and all SG NR levels <31%, 1BwEP-0 step 15 RNO requires going to 1BwFR-H.1 to restore heat sink. Step 1 of 1BwFR-H.1 checks if RCS pressure is above the SG pressure, this is to verify that the SG will act as a heat sink. The answer is no and 1BwFR-C.2 becomes the highest priority functional restoration procedure and will be entered since a transition out of 1BwEP-0 was made and the status trees are applicable.

Task No: S-EP-005

Learn Objective: T.FR03-03

Reference: 1BwEP-0 rev 302, 1BwFR-H.1 rev

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

90

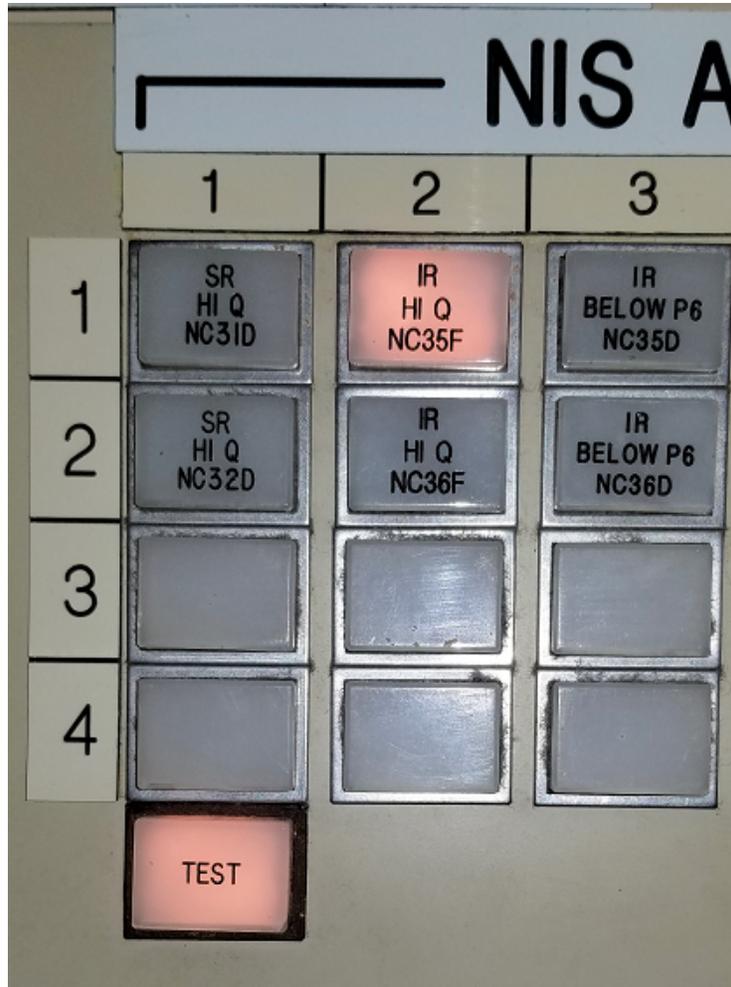
ID: SE10033-N01

Points: 1.00

Unit 1 is at 20% power lowering.

- Power is being reduced to support a MOL maintenance outage per 1BwGP 100-4, POWER DESCENSION.
- Annunciator 1-10-C2, IR CMPSATING VOLT FAILURE, is in alarm.

The pictured indications are noted at NIS AND CONTAINMENT TRIP STATUS LIGHTS TSLB-4:



The US will...

- stabilize and maintain power above P-10 and enter 1BwOA INST-1, NUCLEAR INSTRUMENTATION MALFUNCTION.
- continue the power reduction and manually trip the reactor prior to Mode 2; then perform 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION.
- continue the power reduction and manually trip the reactor in 1BwGP 100-5, PLANT SHUTDOWN AND COOLDOWN.
- direct an immediate manual reactor trip and perform 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION.

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer: A

Answer Explanation
<p>Question: 90 Reference Provided: No History: New for 2018 NRC exam SRO Cog level: High K/A APE33 G2.4.45 -Importance RO 4.1 SRO 4.3 APE033 Loss of Intermediate Range Nuclear Instrumentation G2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm. Tier: 1 Group: 2 K/A match: The question requires the examinee to interpret the bypass permissive panel lights and determine failed intermediate Range NI and prioritize the response. SRO Justification: This question requires the selection and prioritization of procedures to deal with instrumentation failures during a plant shutdown. 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.</p> <p>Answer explanation and plausibility: Choice A is correct. With power at 20% and lowering, P-10 will clear at 10% and an automatic reactor trip will be generated. 1BWOA INST-1 will direct stabilizing the plant to address the instrument failure and allow further power reduction without an automatic reactor trip. Choice B is incorrect. This is incorrect because to get to mode 2, power will have to be lowered below P-10. Lowering power below P-10 would cause a reactor trip with the indicated IR trip status. This is plausible because tripping the reactor before entering the mode of applicability for the IR failure would prevent the automatic reactor trip, changing modes of operation is not the same as changing mode of applicability. Choice C is incorrect. This is incorrect because power will have to be lowered below P-10 to shut the unit down and an automatic reactor trip would occur below P-10 without addressing the failure of the IR first. This is plausible because this would be the path taken once the IR channel failure is bypassed per 1BWOA INST-1; however, based on the indications the IR has not been bypassed in this answer and because of this the reactor will automatically trip once power is lowered below P-10 forcing entry into EP-0. Choice D is incorrect. This is incorrect because failure of the IR NI is not cause for an immediate reactor trip at 20% power. The power descent is for maintenance and can be delayed to complete the actions necessary per 1BWOA INST-1 to bypass the trip and allow an orderly shutdown. This is plausible since the IR NI is required by Tech Spec to reduce power below P-10.</p> <p>Task No: S-AN-001 Learn Objective: 8C.AN-001 Reference: LCO 3.0.4 BwGP 100-4 rev42 draft</p>

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

91

ID: SE10067-N01

Points: 1.00

Both Units are at 100% power.

A malfunction has made CO2 to the 1A DG unavailable, zone 1S-38.

- Automatic detection for the 1A DG, zone 1D-72, is not available.

What actions are the minimum required to maintain the 1A DG available?

- A. ESTABLISH a continuous fire watch and verify backup suppression equipment is available.
- B. ESTABLISH an hourly fire watch and verify backup suppression equipment is available.
- C. ESTABLISH an hourly fire watch and restore the required CO2 in 5 weeks.
- D. ESTABLISH a continuous fire watch and restore the required CO2 in 72 hours.

Answer: C

Answer Explanation

Question: 91

Reference Provided: YES BwAP 1110-1A4 all 9 pages

History: New for 2018 NRC exam SRO Cog level: LOW

K/A APE067AA2.15 -Importance RO 2.9 SRO 3.9

APE067 Plant Fire On Site AA2.15 Ability to determine and interpret the following as they apply to the Plant Fire on Site: Fire Watch Requirements

Tier: 1

Group: 2

K/A match: The question requires the examinee to determine the fire watch requirements to ensure the 1A DG has adequate coverage to ensure any fire is discovered and dealt with before it can cause damage to safety related equipment.

SRO Justification: This question test SRO knowledge of administrative requirements for fire watches to protect systems from fire due to loss of fire detection and suppression.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect, 1S-38 requires an hourly fire watch per condition D.1.1. This is plausible as the first condition listed in BwAP 1110-1A4 has 2 1 hour actions required, establish a continuous fire watch and verify backup suppression is available.

Choice B is incorrect, 1S-38 requires an hourly fire watch per condition D.1.1. This is plausible as D.1.2 is an OR condition to D.1.1 that would allow you to not establish a fire watch if detection was available.

Choice C is correct. 1S-38 requires an hourly fire watch and restore the required CO2 in 5 weeks per condition D.1.1.

Choice D is incorrect, 1S-38 requires an hourly fire watch and restore the required CO2 in 72 hours per condition D.1.1. This is plausible as the B.1.1 is to enter a continuous fire watch.

Task No: R-FP-002

Learn Objective: S.FP1-10

Reference: BwAP 1110-1A4 rev 10

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

92

ID: SE10025-N02

Points: 1.00

Unit 1 is at 100% power.

- During a containment entry, a malfunction in the containment personnel air lock doors resulted in NEITHER door being fully closed.
- 30 minutes later, the interlock was repaired and the doors were closed.
- Unit 1 remains at full power.

In accordance with the Exelon Reportability Manual, this event is reportable to the NRC Operations Center within a maximum of...

- A. 15 minutes.
- B. 1 hour.
- C. 4 hours.
- D. 8 hours.

Answer: D

Answer Explanation

Question: 92

Reference Provided: Yes, LS-AA-1110 rev 25, LS-AA-1120 rev LS-AA-1130 rev 15, LS-MW-1310 rev 007, LS-MW1320 rev 004,LS-MW-1340 rev 017,LLS-AA-1020 rev 26, LS-AA-1010 rev29 collectively called the Exelon Reportability Manual.

History: New for 2018 NRC exam SRO **Cog level:** High

K/A-APE069G2.4.30 Importance RO 2.7 SRO 4.1

APE069 Loss of Containment Integrity G2.4.30 Knowledge of events related to system operation/ status that must be reported to internal organizations or external agencies, such as State, the NRC, or the transmission system operator.

Tier: 1

Group: 2

K/A match: The question requires the examinee to evaluate condition related to loss of containment integrity that require notification to the NRC.

SRO Justification: The question is SRO level because determining event reportability is a function unique to the SRO position.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because a report of a credible security threat requires notification of the NRC within 15 minutes.

Choice B is incorrect. This is plausible because SEC 1.2 Degraded plant condition, interruption of normal operations of a licensed nuclear power reactor through the unauthorized use of or tampering with its machinery, components is a 1 hour report.

Choice C is incorrect. This is plausible SAF 1.2 shutdown notice is 4 hours.

Choice D is correct. SAF 1.4 requires that events that challenge system operability be reported within 8 hours.

Task No: S-AM-102

Learn Objective: 8E.AM-102

Reference: LS-AA-1110 rev 25.

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

93

ID: BWLC3DFR2002

Points: 1.00

1BwFR-C.2, RESPONSE TO DEGRADED CORE COOLING, is in progress at step 11, CHECK IF SI ACCUMULATORS SHOULD BE ISOLATED.

- ALL S/G pressures are 90 psig and stable.
- CETCs are 725°F and slowly rising.

The STA reports that a RED PATH condition on the INTEGRITY Status Tree and 1BwFR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, entry conditions have been met.

The US will ...

- complete 1BwFR-C.2, then transition to 1BwFR-P.1.
- concurrently perform 1BwFR-C.2 and 1BwFR-P.1.
- complete 1BwFR-C.2 through step 12 ONLY, then transition to BwFR-P.1.
- immediately transition to 1BwFR-P.1.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 93

Reference Provided: No

History: LORT Bank question SRO **Cog level:** Memory

K/A-EPE07 EA2.1 Importance RO 3.2 SRO 4.0

EPE007 Saturated Core Cooling EA2.1 Ability to operate and/or monitor the following as they apply to the (saturated core cooling) Facility condition and selection of appropriate procedures during abnormal and emergency operations.

Tier: 1

Group: 2

K/A match: The question requires the examinee to select the appropriate procedure during use of degraded or saturated core cooling.

SRO Justification: This question requires knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is correct. Prior to step 9 of 1BwFR-C.2 is a caution that injecting the SI accumulators will cause an integrity red path and this procedure should be completed before transition to 1BwFR-P.1.

Choice B is incorrect. This is plausible because a red path is a higher priority path than an orange path and completing the actions for core cooling is a priority. It is plausible the US would attempt to run both procedures in parallel. The caution before step 9 of 1BwFR-C.2 clearly states to finish 1BwFR-C.2 before performing 1BwFR-P.1 making this choice incorrect.

Choice C is incorrect. There are procedures such as 1BwEP ES1.3 that have notes stating that action should be completed through a specific step in the procedure before implementing other procedures. This makes this choice plausible. The caution before step 9 of 1BwFR-C.2 clearly states to finish 1BwFR-C.2 before performing 1BwFR-P.1 making this choice incorrect.

Choice D is incorrect. This is plausible because a red path functional restoration procedure takes priority over an orange path procedure. The caution before step 9 of 1BwFR-C.2 clearly states to finish 1BwFR-C.2 before performing 1BwFR-P.1 making this choice incorrect.

Task No: S-FR-009

Learn Objective: 7D.FR-002-B

Reference: 1BwFR-C.2 rev 300

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

94

ID: SG40026-N01

Points: 1.00

The crew is currently staffed with the minimum number of required qualified fire brigade members per BwAP 320-1, SHIFT STAFFING.

- Six hours into a twelve hour shift, a fire brigade member must leave work unexpectedly.

To meet the requirements of BwAP 320-1, the SM/designee...

- A. does NOT need to call out a replacement BECAUSE the Fire Brigade is allowed one unexpected absence for a PARTIAL shift.
- B. does NOT need to call out a replacement BECAUSE the Fire Brigade Chief can fill the member's role during an unexpected absence for a PARTIAL shift.
- C. must take action WITHIN 2 hours to call out a replacement AND have the position filled as soon as possible after the call out.
- D. must take IMMEDIATE action to call out a replacement AND have the position filled within 2 hours.

Answer: D

Answer Explanation

Question: 94

Reference Provided: No

History: Bank question 100 from 2011 NRC exam SRO **Cog level:** Memory

K/A Generic 2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc **-Importance RO 2.9 SRO 3.9**

Tier: 3

Group: 1

K/A match: The questions requires knowledge of minimum staffing requirements for the site fire brigade and the procedural limitations on unavailability.

SRO Justification: This question requires knowledge of limitations in the facility license pertaining to administration of the fire protection system.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because BwAP 320-1 specifically has an exception to allow unexpected absence of fire brigade members. That exception only allows a 2 hour deviation and attempts to restore must begin immediately, making this incorrect.

Choice B is incorrect. This is plausible since the fire brigade chief is a fully qualified fire brigade member and not required per Tech Specs. However, the fire brigade chief is required per the emergency plan, which makes this incorrect.

Choice C is incorrect. This is plausible because BwAP 320-1 specifically has an exception to allow a 2 hour deviation from minimum staffing. The procedure also states attempts to restore must begin immediately, making this incorrect.

Choice D is correct. BwAP 320-1 states that 4 EO members of the fire brigade are required and that the position may be unfilled for 2 hours if call out to restore staffing begins immediately.

Task No: S-AM-029

Learn Objective: 7E.AM-057-A

Reference: BwAP 320-1 rev 25

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

95

ID: SE10040-N01

Points: 1.00

Unit 1 is at 100% power.

One minute later the following indications are noted:

- Annunciator 1-1-A2, CNMT DRAIN LEAK DETECT FLOW HIGH, is in alarm.
- Reactor power is 102% and slowly rising.
- Tave is 585°F and slowly lowering.
- Containment pressure is approximately 2 psig and slowly rising.
- Pressurizer level is 57% and slowly lowering.
- Pressurizer pressure is 2210 psig and slowly lowering.

The US will direct the crew to...

- A. trip the reactor and enter 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, because there is a primary RCS break in containment.
- B. enter 1BwOA PRI-16, RESPONSE TO OVERPOWER CONDITION, and reduce turbine load because there is a dilution event in progress.
- C. enter 1BwOA PRI-1, EXCESSIVE PRIMARY PLANT LEAKAGE, because there is a primary RCS leak in containment.
- D. trip the reactor and enter 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, because there is a secondary steam break in containment.

Answer: D

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 95

Reference Provided: No

History: 2009 NRC exam

SRO Cog level: High

K/A General 2.1.45 Ability to identify and interpret diverse indication to validate the response of another indications. **-Importance** RO 4.3 SRO 4.3

Tier: 3

Group: 1

K/A match: This question requires interpreting diverse indication to validate a secondary break and select appropriate actions.

SRO Justification: This question requires the assessment of plant conditions and the selection of a procedure or section of a procedure to mitigate or recover with which to proceed.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Answer explanation and plausibility:

Choice A is incorrect. This is a plausible choice because the leak is inside containment (cnmt pressure rise) with lowering pressurizer level which is indicative of an RCS break. However, an RCS break would not result in rising reactor power or lowering RCS temperature making this choice incorrect.

Choice B is incorrect. 1BwOA PRI-16 entry conditions are met due to the overpower, making this choice plausible. The event is not a dilution because dilutions would raise RCS temperature instead of lowering it. Also, dilution would not cause severe PZR pressure and level changes making this choice incorrect.

Choice C is incorrect. Lowering pressurizer level and pressure would meet the entry criteria for 1BwOA PRI-1 making this a plausible choice. With addition of lowering temperature and rising reactor power a reactor trip is required making this choice incorrect

Choice D is correct. All indications in combination show a leak of secondary system inside containment. The combination of lowering temperature, pressurizer level and pressure, and rising power indicate energy being released from a secondary break and because containment pressure is rising the leak is not isolable. This requires a reactor trip and entry to 1BwEP-0

Task No: S-EP-001

Learn Objective: 7D.EP-002-D

Reference: 1BwEP-0 rev 302

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

96

ID: SS20038-N01

Points: 1.00

Operating within Tech Spec LCOs ensures that during a steam generator tube rupture, the dose limits of 10 CFR 50.67 are NOT exceeded.

These limits are (1) rem TEDE for an individual at the boundary of the exclusion area for 2 hours OR (2) rem TEDE for an individual in the Main Control Room for the duration of the event.

- A. (1) 50
(2) 25
- B. (1) 50
(2) 5
- C. (1) 25
(2) 25
- D. (1) 25
(2) 5

Answer: D

Answer Explanation

Question: 96

Reference Provided: No

History: Bank used on 2016 NRC exam **SRO Cog level:** Memory

K/A generic 2.2.22 Knowledge of limiting conditions for operations and safety limits Importance
RO 4.0 SRO 4.7

Tier: 3

Group: 2

K/A match: The question meets the K/A, requires examinee knowledge of safety limits found in the bases for LCO 3.4.16 RCS specific activity.

SRO Justification: The question is SRO level because requires knowledge of the Tech Spec bases of LCO 3.4.16.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible, 50 is TODE Federal limit. 25 rem TEDE to any individual at the boundary of the exclusion area. These limits are often confused because of their similar acronyms and overlapping values.

Choice B is incorrect. This is plausible, 50 is TODE Federal limit and 5 rem TEDE for an occupant of the MCR. These limits are often confused because of their similar acronyms and overlapping values.

Choice C is incorrect. This is plausible, 10 CFR 50.67 lists 25 rem TEDE to any individual at the boundary of the exclusion area and 25 REM is the limit for life saving activities. These limits are often confused because of their similar acronyms.

Choice D is correct. 10 CFR 50.67 lists 25 rem TEDE to any individual at the boundary of the exclusion area and 5 rem TEDE for an occupant of the MCR.

Task No: S-TS-006

Learn Objective: 8E.TS-006

Reference: 10CFR 50.67, LCO 3.4.16 bases rev 81 page B 3.4.16-1, ILT lesson plan I1-BZ-XL-01 rev. 3b page 28 and 30

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

97

ID: SG30011-C01

Points: 1.00

Unit 1 is at 100% power.

- A containment entry through the equipment hatch on Unit 1 is required.
- Per BwAP 1450-1, ACCESS TO CONTAINMENT, Attachment 2, CONTAINMENT ENTRY CHECKLIST, the US determines that BwAP 1450-1, Attachment 1, CONTAINMENT ENTRY LOG, is NOT required for the pending entry.

Which of the following is correct justification for NOT requiring the containment entry log?

- A. The equipment hatch door seal trouble alarm is operable.
- B. There are two hatches unlocked and available for emergency egress.
- C. An access control guard is assigned from rad protection.
- D. Surveillance frequency of 1BwOSR 3.6.2.1-1, LLRT OF THE EQUIP HATCH AIRLOCK DOOR GASKET INTERSPACES, is current.

Answer: A

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 97

Reference Provided: No

History: Bank question from 2014 cert exam

SRO Cog level: Memory

K/A Generic 2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

IMPORTANCE RO 3.4 SRO 3.8

Tier: 3

Group: 3

K/A match: This question requires examinee knowledge of containment entry requirements with the Unit at rated power.

SRO Justification: Question is SRO level because the candidate must assess plant conditions and based on detailed knowledge of the procedure, determine log entry (i.e. procedural) requirement.

10CFR55.43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Answer explanation and plausibility:

Choice A is correct. BwAP 1450-1, attachment 2, an entry log is not required if the equipment hatch door seal trouble alarm is operable.

Choice B is incorrect. This is plausible because if two egress points are not available, containment must be evacuated or special conditions met similar to an OA entry and logging might be relaxed until after the event. This does not relax log requirement which is why its incorrect.

Choice C is incorrect. This is plausible as the access control guard would track how many people go in containment logging them in and out. This logging is not related to the official log and entry log requirements of BwAP 1450-1.

Choice D is incorrect. This is plausible as the surveillance tracks LCO requirements for the equipment hatch. Performance of this surv. is required if the alarm is not operable. This is not related to the lack of requirement for a log entry when the alarm is operable.

Task No: S-AM-128

Learn Objective: 7E.AM-128-A

Reference: BwAP 1450-1 rev 44

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

98

Points: 1.00

Unit 2 is at full power.

- A Gas Decay Tank release in progress.

The following AUX BLDG VENT NORM Radiation Monitors indications are received:

- 2PA228 DARK BLUE (Channel operate failure).
- 2PB128 DARK BLUE (Channel operate failure).
- 2PC328 DARK BLUE (Channel operate failure).
- 2PD428 DARK BLUE (Channel operate failure).

To continue the release, the crew is required to ...

- A. initiate grab sampling at least once per 12 hours with the samples analyzed within 24 hours, ONLY.
- B. verify operability of flow loop 0VA019 and restore the rad monitor within 30 days
- C. initiate sampling continuously for ODCM RETS Table 12.4-1, and every 12 hours for noble gas with noble gas samples analyzed within 24 hours and estimate sample flowrate every 4 hours.
- D. initiate an alternate method of monitoring the appropriate parameters within 72 hours, and either restore the monitor to operable status within 7 days or submit a special report within 14 days.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 98

Reference Provided: YES 0BwOS RETS 2.2-1a

History: LORT bank question RO **Cog level:** Memory

K/A G 2.3.15 -Importance RO 3.1 SRO 3.1

G2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey equipment instruments, personnel monitoring equipment, etc.

Tier: 3

Group: 3

K/A match: The question requires the examinee to know the fixed radiation monitoring system and apply that knowledge to interpret and implement the ODCM required actions for a failure.

10CFR55.43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

SRO justification: This question cannot be answered with system knowledge, immediate action or overall mitigative strategy. This question requires understanding of the ODCM manual and its application through 0BwOS RETS 2.2-1a.

Answer explanation and plausibility:

Choice A is incorrect because the continuous monitoring per ODCM requirements and flow rate estimate are also required. This is plausible because the first action required if the release isn't suspended is to perform 12-hour Tiny2018 grab samples.

Choice B is incorrect because the 12-hour grab samples, continuous monitoring per ODCM requirements and flow rate estimate are all required. This is plausible because the next action statement in the 0BwOS RETS 2.2-1a is for plant vent stack flow, and TRM 3.3.i requires flow rate monitoring availability. The flow rate monitor associated with the 2PA228 is not the same flow rate monitor.

Choice C is correct. Per 0BwOS RETS 2.2-1a page 7 the required actions are to initiate 12 hour noble gas grab samples, continuously monitor per ODCM requirements, and estimate the sample flow every 4 hours.

Choice D is incorrect because the continuous monitoring per ODCM requirements and flow rate estimate are also required. This is plausible because the actions for noble gas monitoring channel 1PR01J require verifying real time monitoring of radioactive effluents released via the pathway is to setup and restore the monitor in 7 days or fill out a special report.

Task No: R-AR-001

Learn Objective: S.AR1-15-A

Reference: 0BwOS RETS 2.2-1a rev 11

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

99

ID: SE1WE12-N14-87

Points: 1.00

Unit 1 plant startup from a refuel outage was in progress.

- The crew is currently performing 1BwCA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS.
- Plant conditions have stabilized and the crew is preparing to continue the plant cooldown to cold shutdown conditions.

Prior to initiating the plant cooldown, per 1BwCA-2.1, the crew will contact the TSC to ...

- A. determine the appropriate cooldown method.
- B. determine if SI accumulators should be isolated.
- C. verify adequate shutdown margin.
- D. obtain dose projection of the steam release.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question 99

Reference Provided: NO

History: Bank Question (2014 NRC exam question 87) SRO **Cog Level:** Memory

K/A: Generic 2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Importance: RO 3.8 SRO 4.2

Tier: 3

Group: 4

Meets K/A, examinee must have knowledge of the mitigation strategies of an uncontrolled depressurization of all steam generators, which is a loss of coolant accident, in MODE 3.

SRO Justification: The examinee must assess plant conditions and have detailed knowledge of the procedure to determine the status of the plant and know what actions require coordination with teams outside the main control room.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

Answer: C.

Choice A is incorrect. Plausible because this is done in 1BwEP-3. The only cooling ability at this point would be RH, since four SGs have already cooled the plant down.

Choice B is incorrect. Plausible because in 1BwCA-2.1 accumulators are isolated at step 12.e RNO. The TSC is consulted if there are issues isolating accumulators. This is incorrect since the TSC is not consulted to determine if SI accumulators should be isolated only if they can not be isolated.

Choice C is correct. Step in 1BwCA-2.1 to consult the TSC to verify SDM. This is done with an uncontrolled depressurization of all SGs because there would be concern for SDM since the RCS is cooled so much before substantial boron injection has occurred.

Choice D is incorrect. Plausible because this is done in 1BwEP-1 for a LOCA. Since this is 4 faulted SGs with no SGTR, this is not a concern in 1BwCA-2.1.

Task: S-CA-015

Objective: 7D.CA-004-A

Technical References: 1BwCA-2.1 rev 300 page 34.

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

100

ID: SE10074-C01

Points: 1.00

An inadequate core cooling event is in progress on Unit 1.

- The crew is performing 1BwFR-C.1, RESPONSE TO INADEQUATE CORE COOLING.
- At step 3, CHECK RCP SUPPORT CONDITIONS, support conditions were NOT established.
- Continued attempts to re-establish RCP support conditions have been UNSUCCESSFUL so far.

The crew is now at step 16, CHECK IF RCPs SHOULD BE STARTED.

- ALL RCPs are shutdown.
- CETCs are 1235°F and slowly rising.
- Containment is 2 psig.
- Total AF flow is 600 GPM.
- SG NR levels are as follows:

1A	38%
1B	41%
1C	34%
1D	28%

The US's next direction to the crew will be to...

- A. NOT start an RCP and continue with the next mitigating strategy by opening both PZR PORVs.
- B. remain at step 16 until RCP support conditions are established, then start an RCP.
- C. immediately start an RCP regardless of the RCP support conditions.
- D. transition to SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE.

Answer: C

EXAMINATION ANSWER KEY

2018 NRC SRO Exam

Answer Explanation

Question: 100

Reference Provided: No

History: Bank 2014 Cert Exam

SRO Cog level: High

K/A-Importance: 2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes. **Importance:** RO 3.8 SRO 4.3

Tier: 3

Group: 4

K/A match: The question requires examinee knowledge of operational implication of EOP note during inadequate core cooling

SRO Justification: The question is *SRO level because it requires assessment of conditions and direct appropriate MCR actions and level of detail beyond just the note and overall mitigating strategy.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

The question meets the K/A, requires examinee knowledge of operational implication of EOP note during inadequate core cooling.

Answer explanation and plausibility:

Choice A is incorrect. This is plausible because the RCP is NOT started if SG NR level is less than 31% (adverse containment) then no RCP is started. This is a concern for creep rupture failure mechanism of SG tubes. This is incorrect because SG NR levels are adequate for supporting the start of an RCP.

Choice B is incorrect. This is plausible since RCPs are not normally started without adequate support conditions. Actions of an FR should not be stopped for a potential solution unless directed in the procedure. Core cooling is a higher priority than heat sink and the action should be continued.

Choice C is correct. The note prior to step 16 reads "Normal conditions are desired but NOT required for starting RCPs," a minimum of 10% NR SG level is required to prevent SG tube creep rupture failure and raise the severity of the accident.

Choice D is incorrect. This is plausible because SACRG-1 is entered if CETCs are >1200 F and actions to address the inadequate core cooling are unsuccessful. Not all required actions have been completed in 1BwFR-C.1, so SACRG-1 is not entered at this point.

Task No: S-FR-009

Obj No: 7D.FR-002-A

References: 1BwFR-C.1 rev300