| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 022 K1.01 | |
| | Importance Rating | 3.5 | |

Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS/cooling system

Proposed Question: RO Question # 1

Plant conditions:

- The reactor has tripped.
- RCS pressure is 1820 psig and lowering.
- Containment pressure is 4 psig and rising.
- SG pressures are 1000 psig and stable.
- The crew is performing actions of E-0, Reactor Trip or Safety Injection.

Which ONE of the following describes the Containment Cooling alignment for these conditions?

| | Service Water Outlet Valve FCV-4561 | Service Water Bypass Valve FCV-4562 |
|---|---|---|
| A. | Throttled | Throttled |
| B. | Full Open | Throttled |
| C. | Throttled | Closed |
| D. | Full Open | Full Open |

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plant conditions represent safety injection actuated due to high containment pressure. When SI actuates, Service Water to Containment Coolers goes to full open. Plausible because this is the normal lineup

B. Incorrect. The first part is correct, making it plausible, and also logical that the bypass valve could remain throttled.

C. Incorrect. This is also a potential normal alignment, as the flow control valve would be throttled for temperature control, and it is logical to have the bypass valve closed if not required for flow.

D. Correct. Plant conditions represent safety injection actuated due to high containment pressure. When SI actuates, Service Water to Containment Coolers goes to full open.

Technical Reference(s): 33013-1250, sheet 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R2201C, Obj 1.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 008 K1.02 | |
| | Importance Rating | 3.3 | |

Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: Loads cooled by CCWS

Proposed Question: RO Question # 2

Which ONE of the following describes the component(s) supplied by Component Cooling Water that is (are) isolated by automatic closure of Containment Isolation Motor Operated Valves (MOVs)?

A. Reactor Support Cooling Pad ONLY.

B. Excess Letdown Heat Exchanger ONLY.

C. Non-Regenerative Heat Exchanger ONLY.

D. Reactor Support Cooling Pad, Non-Regenerative Heat Exchanger AND Excess Letdown Heat Exchanger.

Proposed Answer: A

Explanation (Optional):

A. Correct. MOV-813 and MOV-814 receive an automatic isolation signal (T). This isolates Reactor Support Cooling pad only

B. Incorrect. Plausible because the Excess letdown HX has an isolation boundary from the same line as Reactor Support (MOV-817)

C. Incorrect. Plausible because the Non-Regenerative HX is isolated but not located in Containment.

D. Incorrect. Plausible because it identifies all 3 lines in Containment (with exception of RCPs) and it is logical that a Containment Isolation valve would isolate the loads in CTMT

Technical Reference(s): 33013-1246, sheet 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R28091C, Obj 1.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 064 K2.02 | |
| | Importance Rating | 2.8 | |

Knowledge of bus power supplies to the following: Fuel oil pumps

Proposed Question: RO Question # 3

Which ONE of the following describes the power supply to 'A' and 'B' EDG Fuel Oil Transfer Pumps?

A. MCC C and MCC D

B. MCC K and MCC P

C. MCC H and MCC J

D. MCC L and MCC M

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the MCCs are paired in same way as correct MCCs and also because the MCCs are located in Aux Building, where safety related equipment is located

B. Plausible because the MCCs are paired in same way as correct MCCs and also because the MCCs are located in Aux Building, where safety related equipment is located

C. Correct.

D. Plausible because the MCCs are paired in same way as correct MCCs and also because the MCCs are located in Aux Building, where safety related equipment is located

Technical Reference(s): P-12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R0801C, Obj 1.05 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 078 K2.02 | |
| | Importance Rating | 3.3 | |

Knowledge of bus power supplies to the following Emergency air compressor

Proposed Question: RO Question # 4

Plant conditions:

- The plant is at 100% power
- 'C' Instrument Air compressor is running
- All major control systems in AUTO
- Service and Instrument Air are cross-connected per T-1C, Instrument Air/Service Air Cross Connect
- The following alarms are received in the control room:
    - J-29, 480V TRANSFORMER BREAKER TRIP
    - L-22, BUS 15 UNDERVOLTAGE NON-SAFEGUARD

Assuming no action taken by the crew, which ONE of the following describes the Air Compressor that will be running?

A. 'A' Instrument Air Compressor

B. 'B' Instrument Air Compressor

C. 'C' Instrument Air Compressor

D. Service Air Compressor

Proposed Answer: A

Explanation (Optional):

A. Correct. Instrument Air Compressor 'A' is powered from 480VAC Bus 13. It will start on 105 psig air header pressure decreasing

B. Incorrect. Plausible because air compressor is powered from same series of busses but is powered from 480VAC Bus 15.

C. Incorrect. Plausible because air compressor is powered from same series of busses but is powered from 480VAC Bus 15.

D. Incorrect. Plausible because air compressor is powered from same series of busses but is powered from 480VAC Bus 15. The air compressor will not automatically start on loss of another air compressor.

Technical Reference(s): R4701C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R4701C Obj 5.01 (As available)

Question Source: Bank # WTSI 66526

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Ginna 2006 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Editorial modified 2006 question by changing failure to allow for interpretation of alarms

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 026 K3.01 | |
| | Importance Rating | 3.9 | |

Knowledge of the effect that a loss or malfunction of the CSS will have on the following: CCS

Proposed Question: RO Question # 5

Plant conditions:

- The plant is operating at 100% power.
- 'A' Containment Spray pump has been declared INOPERABLE due to an oil leak.
- 'A' and 'D' Containment Recirc Fan Coolers are INOPERABLE and isolated due to service water leaks.
- All other ECCS equipment is OPERABLE.

Can Containment pressure can be maintained within design limits following a LOCA? Why or why not?

A. NO, two (2) Containment Spray pumps must be OPERABLE to meet the design basis for post accident containment cooling and pressure control.

B. NO, if one Containment Spray pump is out of service, four (4) Recirc Fans are required to be OPERABLE to meet the design basis for post accident containment cooling and pressure control.

C. YES, only one (1) OPERABLE Containment Spray pump is required to maintain Containment pressure within limits. Recirc Fan Coolers are not considered for post accident Containment cooling and pressure control.

D. YES, a single OPERABLE Containment Spray pump and two (2) Recirc Fan Coolers OPERABLE meets the design basis for post accident Containment cooling and pressure control.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Only 1 train is required based upon single failure criteria. Plausible because the applicant may assume that only spray provides post accident pressure control

B. Incorrect. Plausible for same reason as option A. If applicant believes that alignment requirements change based upon equipment out of service, then this option may be picked

C. Incorrect. Plausible because pressure control is normally associated with containment spray, while containment coolers are associated with temperature control

D. Correct. One full train of each is required to meet design requirements for post accident cooling and pressure control

Technical Reference(s): ITS Basis 3.6.6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R2201C, Obj 1.06a, b (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Alot of wording changes but left as bank item because conditions are essentially the same

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 076 K3.07 | |
| | Importance Rating | 3.7 | |

Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads

Proposed Question: RO Question # 6

Plant conditions:

- The plant is at 100% power.
- 'B' and 'D' Service Water Pumps are in service.
- The following alarms are received in the Control Room:
  - PPCS-P2160, SERVICE WATER PUMPS A & B HEADER
  - C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW 1050 GPM
- The CO reports that 'B' Service Water Pump has tripped.
- 'A' and 'C' Service Water Pumps tripped upon starting.
- The CRS enters AP-SW.2, Loss of Service Water
- 'D' Service Water Pump is running.
- Service Water Header pressure is 55 psig and stable.

Which ONE of the following describes the action required in accordance with AP-SW.2?

A. Trip the reactor; Align Alternate Cooling Water to 'A' EDG during performance of E-0, Reactor Trip or Safety Injection.

B. Trip the reactor; Align Alternate Cooling Water to 'B' EDG during performance of E-0, Reactor Trip or Safety Injection.

C. Reactor Trip is NOT required; Align Alternate Cooling Water to 'A' EDG and isolate Service Water to Non-Essential loads.

D. Reactor Trip is NOT required; Align Alternate Cooling Water to 'B' EDG and isolate Service Water to Non-Essential loads.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Reactor trip is plausible because it will be performed if there are no SW Pumps operating. In this case, there is one operating. Additionally, Alternate Cooling Water will be supplied to 'A' EDG

B. Incorrect. As in Option A, reactor trip is not required at this time. Alt cooling will be aligned to 'A' EDG, not 'B' EDG

C. Correct. See AP-SW.2

D. Incorrect. Plausible because Alternate Cooling is aligned, but wrong because this option identifies the wrong EDG

Technical Reference(s): AP-SW.2, Rev 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP33C, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 039 K4.06 | |
| | Importance Rating | 3.3 | |

Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following:
Prevent reverse steam flow on steam line break

Proposed Question: RO Question # 7

Plant conditions:

- The plant is in Mode 3.
- RCS temperature is 547°F.
- The Main Condenser is in service.
- A Steam Line Break occurs on 'A' SG inside Containment.

Which ONE of the following describes the plant response to this event?

A. 'A' Main Steam Line Pressure will drop at a higher rate than 'B' Main Steam Line pressure; MSIVs will receive a CLOSE signal on a Containment pressure setpoint of 4 psig.

B. 'A' Main Steam Line Pressure will drop at a higher rate than 'B' Main Steam Line pressure; MSIVs will receive a CLOSE signal on a Containment pressure setpoint of 18 psig.

C. BOTH Main Steam Line pressures will lower at approximately the same rate; MSIVs will receive a CLOSE signal on a Containment pressure setpoint of 4 psig.

D. BOTH Main Steam Line pressures will lower at approximately the same rate; MSIVs will receive a CLOSE signal on a Containment pressure setpoint of 18 psig.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the steam pressure response is correct and because there are 2 other MSLI signals that are dependent on SI being generated, which is at 4 psig. These signals are coincident with Low Tavg

B. Correct. Containment pressure at 18 psig (HIGH-2) will initiate MSIV closure. One SG

stays constant because there is a check valve downstream of the MSIV that will close on slight pressure differences between steam lines, and in this case, with the Main Condenser in service, the steam lines are connected

C. Incorrect. Plausible because if only the MSIV closure were taken into consideration and not the check valve, and the MSIV closure signals associated with SI, this option would be chosen.

D. Incorrect. Plausible because the setpoint is correct and because the applicant may not consider that there is a check valve prior to MSLI

Technical Reference(s): R4001C, Rev 23 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R4001C, Obj 1.04, 1.07 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 013 K4.08 | |
| | Importance Rating | 3.1 | |

Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following Redundancy

Proposed Question: RO Question # 8

Which ONE of the following describes an Engineered Safety Feature as defined by Technical Specifications, and the logic required for actuation of the feature?

A. Containment Pressure High; 2 out of 3

B. Containment Pressure High; 2 out of 4

C. Pressurizer Pressure High; 2 out of 3

D. Pressurizer Pressure High; 2 out of 4

Proposed Answer: A

Explanation (Optional):

A. Correct. Containment Pressure High is an ESFAS instrument that will initiate Safety Injection and Containment Isolation.

B. Incorrect. Plausible because the parameter is correct, Logic is plausible because 2 of 4 is a standard logic.

C. Incorrect. PRZR High pressure is am RPS instrument, but plausible because PRZR low pressure is RPS as well as ESFAS. Logic is correct.

D. Incorrect. Same description as C, but logic is incorrect and plausible as in option B

Technical Reference(s): TS 3.3.2
AR-A-28 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3501C, Obj 1.13 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 004 K5.44 | |
| | Importance Rating | 3.2 | |

Knowledge of the operational implications of the following concepts as they apply to the CVCS: Pressure response in PZR during in-and-out surge

Proposed Question: RO Question # 9

Plant conditions:

- The plant is at 100% power.
- Control Rods are in MANUAL.
- An EH System malfunction caused a load rejection of approximately 50 MWe.

Which ONE of the following describes the immediate effect on pressurizer pressure and Charging flow?

| | Pressurizer Pressure | Charging Flow |
|---|---|---|
| A. | RISES | RISES |
| B. | RISES | LOWERS |
| C. | LOWERS | RISES |
| D. | LOWERS | LOWERS |

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Pressurizer pressure rises because RCS temperature rises during the load rejection. RCS water is forced into the pressurizer (INSURGE), causing the bubble to be squeezed. The water forced into the pressurizer causes level to rise above program. Charging flow is automatically reduced to bring level back to program

B. Correct. As described in Option A above

C. Incorrect. Opposite of actual effect. After the immediate response of pressurizer pressure, the colder RCS water will tend to depressurize the saturated pressurizer. This is why heaters are turned on for a high level deviation. If applicant considers this

phenomenon, they will choose this option

D. Incorrect. Charging flow does immediately lower, but pressure doesn't lower until later in the event.

Technical Reference(s): ROC02C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: ROC02C Obj (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

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.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 006 K5.01 | |
| | Importance Rating | 2.8 | |

Knowledge of the operational implications of the following concepts as they apply to ECCS: Effects of temperatures on water level indications

Proposed Question: RO Question # 10

Which ONE of the following describes the parameters affected by adverse containment values when performing the SI flow reduction sequence in ES-1.2?

A. Pressurizer Level ONLY

B. RCS Subcooling ONLY

C. RCS Pressure AND Pressurizer Level

D. Pressurizer Level AND RCS Subcooling

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Answer is correct but not complete. RCS Subcooling is also considered

B. Incorrect. Answer is correct but not complete. Pressurizer Level is also considered

C. Incorrect. RCS pressure is plausible because it has an adverse value for stopping RHR pumps earlier in the procedure

D. Correct. Both parameters are higher for adverse values than normal values. This is based on either radiation levels or pressure in containment, with pressure causing high atmospheric temperatures that cause inaccurate instrument readings

Technical Reference(s): ES-1.2, Rev 33, Step 12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RES12C, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 012 K6.03 | |
| | Importance Rating | 3.1 | |

Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Trip logic circuits

Proposed Question: RO Question # 11

The plant is at 100% power.

PRZR Pressure Channel 429 has failed and has been properly defeated in accordance with ER-INST.1, Reactor Protection Bistable Defeat After Instrumentation Loop Failure.

Which ONE of the following identifies the Reactor Trip and Safety Injection actuation logic required (from the remaining in-service channels) on Low PRZR Pressure?

| | Reactor Trip | Safety Injection |
|---|---|---|
| A. | 1/3 | 1/2 |
| B. | 2/3 | 1/2 |
| C. | 1/3 | 2/3 |
| D. | 2/3 | 2/3 |

Proposed Answer: A

Explanation (Optional):

A. Correct. Reactor trip is normally 2 of 4 logic. The failed channel will be placed in trip and one additional channel in trip will initiate a reactor trip. Normally SI 2 of 3 on Low PRZR Pressure, now 1 of 3

B. Incorrect. Plausible because P-11 and PRZR level reactor trips employ normal 2 of 3 logic. Also because SI logic is correct.

C. Incorrect. Plausible because applicant may not understand that ESF functions are placed in trip condition, they may consider the channel bypassed, not tripped.

D. Incorrect. If channels were placed in bypass instead of trip, this would be correct for RPS. For SI, it is incorrect but plausible because normal logic is 2 of 4. SI is 2 of 3 on low PRZR pressure

Technical Reference(s): P-1, Rx Control and Protection (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3501C, Obj 1.06 (As available)

Question Source: Bank # WTSI 59465

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: McGuire 2008

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 003 K6.02 | |
| | Importance Rating | 2.7 | |

Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: RCP seals and seal water supply

Proposed Question: RO Question # 12

Plant Conditions:

- Reactor power is stable at 100% when the following alarm actuated:
  - A-17, RCP 1A NO. 1 SEAL HI-LO FLOW 5.0 GPM 1.0
  - B-3, RCP A STANDPIPE HI LEVEL +1 FT
- "A" RCP #1 seal leakoff is 0.9 gpm and steady
- The Head Control Operator calculated the leak rate to the RCDT to be 2.0 gpm and steady.
- The Crew entered AP-RCP.1, RCP SEAL MALFUNCTION

Which of the following are required actions per AP-RCP.1?

A. Secure the "A" RCP in 8 hours.

B. Trip the reactor and "A" RCP and close #1 seal leakoff isolation valve after 4 minutes.

C. Continue operation and increase surveillance frequency of the "A" RCP parameters.

D. Be in MODE 3 in 6 hours and MODE 5 in 36 hours.

Proposed Answer: A

C also correct, per revised key. ✓ fish

Explanation (Optional):

A. Correct. Total #1 Seal Flow is defined as the sum of #1 Seal Leakoff Flow and #2 seal leak rate to the RCDT.

B. Incorrect. Plausible because the actions would be correct per AP-RCP.1 if the Total #1 Seal Flow from the "A" RCP was greater than 8 GPM.

C. Incorrect. Plausible because the actions would be correct if the #3 seal failed vice the number #1 seal.

but there is not any requirement to be in MODE 5 in 36 hours.

Technical Reference(s): AP-RCP.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R1301C, Obj 2.02 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 007 A1.02 | |
| | Importance Rating | 2.7 | |

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining quench tank pressure

Proposed Question: RO Question # 13

Plant conditions:

- The plant is at 100% power.
- The following annunciator is received:
  - F-9, PRT HI PRESS 5 PSI
- PRT pressure is 6 psig and RISING SLOWLY.
- PRT level is 64% and STABLE.

If allowed to continue, ...

The PRT rupture disc will discharge to containment when pressure rises to__(1)__.

The crew must __(2)__ to prevent PRT rupture disc operation.

A. (1) 50 psig
(2) vent the PRT

B. (1) 50 psig
(2) drain the PRT to the RCDT

C. (1)100 psig
(2) drain the PRT to the RCDT

D. (1)100 psig
(2) vent the PRT

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because action is correct but setpoint for PRT rupture disc is incorrect.

B. Incorrect. Plausible because action would be taken IAW the AR, but level is within the low end of the normal band, so that action would not be required. (Low level is 60.8%)

C. Incorrect. Rupture disc pressure is correct but action is incorrect as in option B

D. Correct. Rupture disc blows at 100 psid, and for high pressure without high level, venting the PRT using AOV-527 is the correct action

Technical Reference(s): AR-F-9
P-2, RCS P & L (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R1401C, Obj 1.07 (As available)

Question Source: Bank # WTSI 56080
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Wolf Creek 2007

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 063 A1.01 | |
| | Importance Rating | 2.5 | |

Ability to predict and/or monitor changes in parameters associated with operating the dc electrical system controls including: Battery capacity as it is affected by discharge rate

Proposed Question: RO Question # 14

Plant conditions:

- A loss of all AC power has occurred.
- The crew is performing actions of ECA-0.0, Loss of All AC Power.
- The crew is evaluating load shed of the Batteries

Which ONE of the following describes the reason for requirement to shed non-essential DC loads in accordance with ECA-0.0, Loss of All AC Power?

A. Battery discharge rate is reduced to ensure the station meets the 2 hour technical specification design basis requirement for battery capacity following a loss of AC power.

B. Battery discharge rate is reduced to ensure the station meets the 4 hour technical specification design basis requirement for battery capacity following a loss of AC power.

C. Battery discharge rate is reduced to ensure the station meets the 2 hour coping requirement for loss of all AC power

D. Battery discharge rate is reduced to ensure the station meets the 4 hour coping requirement for loss of all AC power

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible since there is a design basis assumption contained in TS, but it is 4 hours, not 2 hours. 2 hours is plausible because it is the allowed TS action time for loss of DC.

B. Incorrect. Plausible since the TS design basis is 4 hours, but load shedding is not required to achieve design basis operation of the battery

C. Incorrect. In accordance with the Station Blackout Program Plan, the coping requirement is 4 hours. 2 Hours is plausible because of the TS action time in section 3.8 for battery or DC bus inoperability

D. Correct. Load is shed to ensure that a loss of AC power lasting up to 4 hours will not fully discharge the batteries

Technical Reference(s): ECA-0.0 step 17 and background document
Station Blackout Program Plan (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: REC00C, Obj 1.03 (As available)

Question Source: Bank #
Modified Bank # WTSI 19213 (Note changes or attach parent)
New

Question History: Last NRC Exam: Harris 2005 NRC

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 005 A2.01 | |
| | Importance Rating | 2.7 | |

Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure modes for pressure, flow, pump motor amps, motor
temperature, and tank level instrumentation

Proposed Question: RO Question # 15

Plant conditions:

- The plant is in Mode 5.
- Reduced Inventory Operations are in progress for work on Reactor Coolant Pumps.
- 'A' RHR pump is operating.
- RHR flow was initially at 450 GPM.
- RHR Discharge flow is currently oscillating between 50 and 800 GPM.
- Pump motor amps are oscillating, and discharge pressure is fluctuating.
- RCS temperature is 110°F and slowly rising.
- The CRS enters AP-RHR.2, Loss of RHR While Operating at Reduced Inventory Conditions.

Which ONE of the following is occurring, and which of the following actions is required for operation of RHR Pumps in accordance with AP-RHR.2?

A. Air entrainment in RHR suction; Reduce RHR flow with RHR Pump 'A' running until conditions stabilize.

B. Air entrainment in RHR suction; Stop 'A' RHR Pump; After restoring RCS conditions to support operation of an RHR Pump, restart one RHR Pump.

C. RHR Pump 'A' has reached Runout conditions; Reduce RHR flow with RHR Pump 'A' running until conditions stabilize.

D. RHR Pump 'A' has reached Runout conditions; Stop RHR Pump 'A'; After restoring RCS conditions to support operation of an RHR Pump, restart one RHR Pump.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Action would be correct for normal cavitation, but at reduced RCS level, just reducing flow will not alleviate air entrainment

B. Correct. Discharge pressure and flow fluctuations indicate that NPSH is lost. This is caused by low level and high flow for that level. Action is correct in accordance with philosophy of AP-RHR.2. Air is vented from the system prior to restart of RHR pumps

C. Incorrect. Plausible because actions are reasonable for the condition and prescribed by the procedure for flow being 2 high for existing level, and because high flow is consistent with Runout (Fluctuations to 800 GPM)

D. Incorrect. Plausible because actions are correct for the actual condition presented, and because high flow is consistent with Runout (Fluctuations to 800 GPM)

Technical Reference(s): AP-RHR.2
Westinghouse ARG-1
Background, Loss of RHR at Mid-Loop conditions (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP25C, Obj 2.01 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 010 A2.01 | |
| | Importance Rating | 3.3 | |

Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater failures

Proposed Question: RO Question # 16

Plant conditions:

- The plant is at 100% power when a plant transient occurred.
- After the transient, power is 96% and stable.
- The following alarms are received in the control room:
  - F-6, PRESSURIZER HEATER BREAKER TRIP
  - F-10, PRESSURIZER LO PRESS 2205 PSI
- PRZR pressure indicates 2200 psig and slowly lowering.
- The HCO reports that PRZR Backup Heater breaker has tripped.
- The CRS enters AP-PRZR.1, Abnormal Pressurizer Pressure.

Based on the above, entry into Technical Specification is ___(1)____ and the action required to restore backup heaters is ___(2)___...

A. (1) REQUIRED
(2) reset the breaker, place to ON, and verify load increase on Bus 14.

B. (1) REQUIRED
(2) reset the breaker, place to ON, and verify load increase on Bus 16.

C. (1) NOT REQUIRED
(2) reset the breaker, place to ON, and verify load increase on Bus 14.

D. (1) NOT REQUIRED
(2) reset the breaker, place to ON, and verify load increase on Bus 16.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because everything is correct with the exception of bus 14 loading

B. Correct. LCO for RCS pressure must be entered (DNBR, 3.4.1) Action is correct, and pressurizer backup heaters are fed from Bus 16

C. Incorrect. Plausible because the applicant may only consider TS 3.4.9 for pressurizer heater capacity. In this case, the DNBR TS LCO must be addressed. Additionally, Bus 14 is wrong

D. Incorrect. Plausible because the applicant may only consider TS 3.4.9 for pressurizer heater capacity. In this case, the DNBR TS LCO must be addressed.

Technical Reference(s): TS 3.4.1
COLR
AP-PRZR.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP11C, Obj 2.01 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 059 A3.06 | |
| | Importance Rating | 3.2 | |

Ability to monitor automatic operation of the MFW, including: Feedwater isolation

Proposed Question: RO Question # 17

Plant conditions:

- A plant shutdown is in progress.
- Reactor power is at approximately 60%.
- A failure of ADFCS for 'B' Steam Generator causes the 'B' Feedwater Regulating Valve to remain in its current position.

Assuming NO action by the crew, which ONE of the following describes the effect on the plant as turbine load is reduced?

A. 'B' SG level will rise and a Feedwater Isolation signal will be sent to 'B' SG ONLY at 85% narrow range level

B. 'B' SG level will rise and a Feedwater Isolation signal will be sent to BOTH SGs when 'B' SG reaches 85% narrow range level

C. 'B' SG level will lower and a reactor trip will occur when 'B' SG level reaches 20%

D. 'B' SG level will lower and a reactor trip will occur when 'B' SG level reaches 17%

Proposed Answer: A

Explanation (Optional):

A. Correct. If feedwater requirements are lowering with the feedwater reg valve in its current position, overfeeding will occur at some point during the shutdown.

B. Incorrect. Feedwater isolation occurs only for the affected SG. Results in closing Feedwater Reg Valves and Feedwater Reg Bypass Valves

C. Incorrect. Plausible because SG level will change but it will go in the opposite direction. The concept of a valve failing in position is easily confused. This setpoint is incorrect,

but plausible because it is significantly below normal level, and some APs require a trip at this setpoint

D. Incorrect. Plausible because SG level will change but it will go in the opposite direction. The concept of a valve failing in position is easily confused. Setpoint is correct however

Technical Reference(s): 33013-1352, Sheet 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R4401C, Obj 1.06 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 103 A3.01 | |
| | Importance Rating | 3.9 | |

Ability to monitor automatic operation of the contain-ment system, including: Containment isolation

Proposed Question: RO Question # 18

Plant conditions:

An RCS leak resulted in the following conditions:

| TIME | EVENT |
|---|---|
| 0812 | Manual Reactor Trip. |
| 0826 | Pressurizer Pressure 1800 psig and lowering. |
| 0828 | Manual Safety Injection. |
| 0907 | Containment Pressure 4 psig and rising. |
| 0941 | Containment Pressure 29 psig and rising. |
| 1003 | RCS Pressure 220 psig and stable. |

Assuming NO additional actions were taken, which ONE of the following choices describes the EARLIEST time an AUTOMATIC Containment Isolation signal was generated?

A. 0826

B. 0828

C. 0907

D. 0941

Proposed Answer: C

Explanation (Optional):

A. Incorrect. PRZR pressure is not quite below the SI setpoint yet. Plausible because it is close to setpoint.

B. Incorrect. Manual SI does not initiate Containment isolation. Plausible because CI is associated with SI actuation, and does actuate on any other auto SI

C. Correct. Containment pressure of 4 psig will initiate Si and CI will be actuated based on that signal

D. Incorrect. Plausible because setpoint for Containment Spray is 28 psig, which is associated with containment pressure HIGH-3 signal

Technical Reference(s): P-1, Rx Control and Protection (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3501C, Obj 1.07 (As available)

Question Source: Bank #

Modified Bank # WTSI 66192 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Modified times and shifted events to arrive at different answer from Ginna 2007

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 062 A4.04 | |
| | Importance Rating | 2.6 | |

Ability to manually operate and/or monitor in the control room: Local operation of breakers

Proposed Question: RO Question # 19

Plant conditions:

- The plant is in Mode 5 with the 'A' Residual Heat Removal Pump in service.
- During a plant heatup the control power fuse for the 'A' Residual Heat Removal Pump blows.

Which ONE of the following describes how the Main Control Board Residual Heat Removal Pump indication and local breaker control is affected by the loss of control power?

A. Main Control Board red / green running indications will be lost.
Local OPEN / CLOSE light indication is available, and local breaker control will be lost until control power is restored.

B. Main Control Board red / green running indications will be lost.
Local OPEN / CLOSE mechanical indication is available, and local breaker control is possible without the control power.

C. Main Control Board red / green running indications will be available.
Local OPEN / CLOSE light indication is available, and local breaker control is possible without the control power.

D. Main Control Board red / green running indications will be available.
Local OPEN / CLOSE mechanical indication is available, and local breaker control will be lost until control power is restored.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because Main Control Board red / green running indications will be lost, however, local breaker control is possible without control power. The breaker may be mechanically operated locally

B. Correct. With a loss of control power, Main Control Board red / green running indications will be lost. Local breaker control is still possible.

C. Incorrect. Plausible because local breaker control is possible without control power, however, local OPEN / CLOSE light indication is NOT available.

D. Incorrect. Plausible because local OPEN/CLOSE indication is available, however, Main control board indications are lost and local control is available.

Technical Reference(s): LP RGF11C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RGF11C, Obj (As available)

Question Source: Bank # WTSI 63033

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Comanche Peak 2009

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 073 A4.02 | |
| | Importance Rating | 3.7 | |

Ability to manually operate and/or monitor in the control room Radiation monitoring system control panel

Proposed Question: RO Question # 20

Plant conditions:

- R-5 indicates a RANGE Alarm
- Detector Display indicates 0.00
- Monitor Bar Graph display is extinguished

Which ONE of the following describes the reason for this indication, and the condition required to reset the alarm?

A. Detector power loss; alarm must be manually reset once the condition is clear

B. Detector power loss; alarm will automatically reset if the condition clears

C. Radiation field is below the instrument range; alarm will automatically reset once the condition is clear

D. Radiation field is above the instrument range; alarm must be manually reset once the condition is clear

Proposed Answer: C

Explanation (Optional):

A. Incorrect. If the detector had a power loss, the display would read low. However, this condition would cause a FAIL alarm.

B. Incorrect. Power loss will also cause a fail alarm, but it is correct that auto reset occurs

C. Correct. A RANGE alarm with display of all zeroes indicates that the instrument is below minimum range

D. Incorrect. Plausible because there is an E-Value for over-range, (EEEEE) but if the detector was over-ranged, it would automatically reset once the detector returned to

scale

Technical Reference(s): R3901C, Rev 23 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3901C, Obj 1.07 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 061 2.4.34 | |
| | Importance Rating | 4.2 | |

Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: RO Question # 21

Plant conditions:

- The reactor was manually tripped in accordance with AP-CR.1, Control Room Inaccessibility.
- The Control Room has been evacuated due to a toxic atmosphere.
- There is NO fire in progress.

Which ONE of the following describes the method for local operation of Auxiliary Feedwater?

A. MDAFW Pumps throttled to maintain approximately 52% Narrow Range SG level

B. MDAFW Pumps throttled to maintain approximately 350 inches Wide Range SG level

C. SAFW Pump throttled to maintain approximately 52% Narrow Range SG level

D. SAFW Pump throttled to maintain approximately 350 inches Wide Range SG level

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because this is the normal level maintained in SGs. Also plausible because MDAFW is preferred

B. Correct. See procedure step 6 performed by HCO

C. Incorrect. SAFW pump is plausible, because it is an alternate means of providing AFW and the pump also has local controls to start if required. NR level plausible because it is the normal level maintained

D. Incorrect. Same reason as C and also because WR level is correct

Technical Reference(s): AP-CR.1, Rev. 24
R5101C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP04C, Obj 2.01 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 076 2.1.27 | |
| | Importance Rating | 3.9 | |

Conduct of Operations: Knowledge of system purpose and / or function.

Proposed Question: RO Question # 22

Plant conditions:

- The plant is at 100% power.
- 'A' and 'D' Service Water Pumps are running.
- 'B' and 'C' Service Water Pumps are selected.
- Safety Injection occurs.
- Immediately following the Safety Injection actuation, a loss of off-site power occurs.

Which ONE of the following describes (1) the number of Service Water Pumps that will be running in <u>each</u> train, and (2) a safety related load that will be automatically isolated?

A. (1) ONE
(2) AFW Pump thrust bearing and oil coolers

B. (1) ONE
(2) CCW Heat Exchangers

C. (1) TWO
(2) AFW Pump thrust bearing and oil coolers

D. (1) TWO
(2) CCW Heat Exchangers

Proposed Answer: B

Explanation (Optional):

A. Incorrect. One pump is selected in each train and will be running upon completion of the sequencing, because the running (but not selected) pump is stripped (load shed).

B. Correct. One pump will start (selected) in each train. Non-critical load is also correct

C. Incorrect. Number of pumps is incorrect but plausible because if this was SI only, both

pumps would be running

D. Incorrect. Plausible because the load is correct, and if off-site power was available, both pumps would be running

Technical Reference(s): R5101C, Rev 27 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R5101C, Obj 1.02 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Meets KA because knowledge of the purpose of the system during ESFAS actuation is required to answer the question

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 013 K3.01 | |
| | Importance Rating | 4.4 | |

Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Fuel

Proposed Question: RO Question # 23

Plant conditions:

- A LOCA has occurred.
- RCS pressure is 100 psig.
- SI Pumps 'A' and 'B' are tripped.
- 'C' SI Pump is running.
- RHR Pump 'A' and 'B' are tripped.
- RVLIS indicates Rx Vessel level is 20%.

If this condition continues, which ONE of the following describes the effect on the fuel assemblies?

A. Fuel failure is NOT likely to occur. Minimum safety function requirements are met.

B. Fuel failure is NOT likely to occur. SI Accumulator injection will maintain core cooling.

C. Fuel failure is likely to occur. Minimum safety function train requirements are met, but RVLIS indication is too low to sustain core cooling.

D. Fuel failure is likely to occur. Minimum safety function train requirements are NOT met and RVLIS indication is too low to sustain core cooling.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Only one SI pump running with RHR pump flow significantly below what it should be, at least one train of ECCS is not available or running properly.

B. Incorrect. Accumulators should have already injected and RVLIS indication shows that Reactor Vessel level is low.

C. Incorrect. Plausible because 'C' SI Pump is running, but wrong because it is not

providing enough flow to sustain core cooling since no other equipment is running, and it is true that RVLIS is low.

D. Correct. Minimum safety function requirements of at least one full train of ECCs is NOT met at this time, and fuel damage is likely to occur

Technical Reference(s): UFSAR 15.6.4.1.3.2
Westinghouse Setpoint Document (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: N/A (As available)

Question Source: Bank # WTSI 52628
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: VC Summer 2006

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

10 CFR Part 55 Content: 55.41 8
55.43

Components, capacity, and functions of emergency systems.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 010 2.4.20 | |
| | Importance Rating | 3.8 | |

Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes.

Proposed Question: RO Question # 24

Plant conditions:

- A Steam Generator Tube Rupture has occurred.
- Off-Site power was lost shortly after the reactor was tripped.
- 'A' Steam Generator is isolated.
- RCS cooldown is in progress.
- The crew is preparing to depressurize the RCS to refill the pressurizer.

Which ONE of the following is a concern associated with the depressurization for these conditions, in accordance with E-3, Steam Generator Tube Rupture?

A. Thermal shock to the pressurizer spray nozzle

B. Loss of RCS pressure control when the PRZR goes solid

C. Voiding in the upper head region of the RCS

D. Voiding in the RCS side of the tube bundle region of the ruptured SG

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Since off-site power is lost, RCPs will be tripped, causing normal spray to be lost. PORVs are the second choice to depressurize. Plausible because this may be a concern if aux spray had to be used

B. Incorrect. Using the PORVs will cause PRZR level to rapidly rise, and the depressurization may need to be terminated in High PRZR level, but the reason for that would be voiding under the head. pressure control would not be lost

C. Correct. See caution prior to step 22. With no RCPs running, there is no circulation in

the head area. Depressurization may result in voiding

D. Incorrect. Voiding should not be occurring in tube region because the SG is isolated and tube bundle is covered with water

Technical Reference(s): E-3, caution for step 22 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: REP03C, Obj 1.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 007 A3.01 | |
| | Importance Rating | 2.7 | |

Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT

Proposed Question: RO Question # 25

Plant conditions:

- The plant is at 100% power.
- All control systems are operating normally in AUTO.
- Letdown Pressure transmitter PT-135 fails low.

Which ONE of the following describes how Letdown Pressure Control Valve, PCV-135, responds and which tank level will increase as a result of this failure?

A. PCV-135 closes; RCDT level will increase.

B. PCV-135 closes; PRT level will increase.

C. PCV-135 opens; RCDT level will increase.

D. PCV-135 opens; PRT level will increase.

Proposed Answer: B

Explanation (Optional):

A. Plausible because the RCDT is the only other tank in Ctmt that could receive water from the primary system.

B. Correct. If pressure input fails low, the valve will close in an attempt to maintain pressure. When the valve closes, Letdown is essentially isolated, and the LP relief valve to the PRT (203) will lift.

C. Plausible because the valve is reverse acting, which provides a common misconception on valve fail position. VCT and LRW are the 2 flow paths of water that are possible if the valve did fail open. VCT normally and Waste Holdup Tank if VCT level exceeded the high setpoint for divert.

D. Plausible because the valve is reverse acting, which provides a common misconception on valve fail position. VCT and LRW are the 2 flow paths of water that are possible if the valve did fail open. VCT normally and Waste Holdup Tank if VCT level exceeded the high setpoint for divert.

Technical Reference(s): R1601C, Rev 24
CPI-PRESS-135 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R1601C, Obj 1.07g, 1.10j (As available)

Question Source: Bank # WTSI 63653
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Harris 2007 NRC

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 039 A2.05 | |
| | Importance Rating | 3.3 | |

Ability to (a) predict the impacts of the following mal-functions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Increasing steam demand, its relationship to increases in reactor power

Proposed Question: RO Question # 26

Plant conditions:

- Reactor Startup is in progress following a mid-cycle outage.
- The plant is at 5% power when a SG ARV fails open.
- RCS temperature decreases and stabilizes at 538°F.

Which of the following predicts the plant response and the operator actions required in accordance with plant procedures and Technical Specifications?

A. Reactor power rises; Restore Tave above the Technical Specification minimum or the unit must be subcritical in Mode 2 in a maximum time of 15 minutes

B. Reactor power rises; Restore Tave above the Technical Specification minimum or the unit must be subcritical in Mode 2 in a maximum time of 30 minutes

C. The reactor becomes subcritical; withdraw control rods to raise Tavg above the Technical Specification minimum within a maximum of 15 minutes or initiate a shutdown to Mode 3

D. The reactor becomes subcritical; withdraw control rods to raise Tavg above the Technical Specification minimum within a maximum of 30 minutes or initiate a shutdown to Mode 3

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible since reactor power will increase, but temperature is not to be restored within 15 minutes. 30 minutes is allowed in accordance with TS 3.4.2.

B. Correct. Temperature may then be recovered by using control rods in a slow and controlled manner. Temperature has to be restored to greater than 540°F within 30 minutes due to the requirements of TS 3.4.2

C. Incorrect. Plausible since the 15 minute time limit is similar to the time associated with restoration, but the reactor does not become subcritical.

D. Incorrect. Plausible since the time is correct, but the reactor does not become subcritical.

Technical Reference(s): TS 3.4.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

Question Source: Bank # WTSI 19193

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Harris 2005

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 063 A4.01 | |
| | Importance Rating | 2.8 | |

Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses

Proposed Question: RO Question # 27

During operation at power with the Reactor Trip Breakers (RTBs) closed, a loss of 125 VDC control power to one of the RTBs occurs.

Which ONE of the following describes the effect on the RTB?

A. RTB opens due to loss of power to the undervoltage trip coil.

B. RTB opens due to loss of power to the shunt trip coil.

C. RTB remains closed, and the undervoltage trip coil will not function on a reactor trip signal from the Reactor Protection System.

D. RTB remains closed, and the shunt trip coil will not open on a reactor trip signal from the Reactor Protection System.

Proposed Answer: A

Explanation (Optional):

A. Correct. Indication, UV coil, and shunt trip coil receive power from DC bus. Loss of DC results in a loss of power to the UV coil, causing it to drop out and causing the breaker to open.

B. Incorrect. The breaker does not trip on loss of power to shunt coil because the shunt coil requires control power to operate. The undervoltage coil losing power would cause a reactor trip breaker to open.

C. Incorrect. Indication is lost and power is lost to UV coil, but plausible because the breaker shunt coil uses control power and it will not be capable of tripping on a shunt trip.

D. Incorrect. Indication and shunt trip capability lost. Plausibility is as described in options above.

Technical Reference(s): R3501C Rev 28 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3501C, Obj 1.10 (As available)

Question Source: Bank # WTSI 48308

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Callaway 2005

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 003 K5.02 | |
| | Importance Rating | 2.8 | |

Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP coastdown on RCS parameters

Proposed Question: RO Question # 28

The plant is at 100% power:

- RCP 'B' trips due to a failure of the RCP breaker.
- The reactor trips.

Which ONE of the following would be CORRECT regarding the loop 'B' SG level and RCS Loop Delta T INITIAL responses following the RCP trip?

A. 'B' SG level would shrink lower than 'A' SG immediately after the RCP trips.
RCS loop 'B' delta T would be lower than RCS loop 'A' delta T

B. 'B' SG level would swell higher than 'A' SG immediately after the RCP trips.
RCS loop 'B' delta T would be lower than RCS loop 'A' delta T

C. 'B' SG level would swell higher than 'A' SG immediately after the RCP trips.
RCS loop 'B' delta T would be higher than RCS loop 'A' delta T

D. 'B' SG level would shrink lower than 'A' SG immediately after the RCP trips.
RCS loop 'B' delta T would be higher than RCS loop 'A' delta T

Proposed Answer: A

Explanation (Optional):

A. Correct. SG level will shrink immediately following the RCP trip due to less heat input. Loop delta T will lower due to no heat removal through the affected loop, as well as reverse flow since the other loop is still in service

B. Incorrect. Plausible candidate could think SG level would swell due to no heat removal from the loop or that feedwater will cause a rise. Delta T response is correct

C. Incorrect. Plausible because candidate could think SG level would swell due to no heat removal from the loop and think delta T could rise due to setting up natural circulation.

In this case, natural circulation would not occur because the other loop is in service

D. Incorrect. SG level response is correct. Plausible candidate could think no heat removal from the loop could cause a higher delta T, or consider natural circulation setting up

Technical Reference(s): LP ROC01S (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: ROC01S, Obj (As available)

Question Source: Bank # WTSI 64678

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Vogtle 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

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.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 011 K3.01 | |
| | Importance Rating | 3.2 | |

Knowledge of the effect that a loss or malfunction of the PZR LCS will have on the following: CVCS

Proposed Question: RO Question # 29

Plant Conditions:

- Reactor power = 90% and stable
- Pressurizer level = 52% and stable
- "A" charging pump is running in AUTO
- "C" charging pump is running in Manual
- The Average Tavg input to pressurizer level has failed low

Assuming no operator action, how will (1) the "A" charging pump respond and (2) what will be the status of the pressurizer backup heaters?

A. (1) "A" charging pump will slow down
(2) Backup heaters will be energized

B. (1) "A" charging pump will speed up
(2) Backup heaters will be energized

C. (1) "A" charging pump will slow down
(2) Backup heaters will be deenergized

D. (1) "A" charging pump will speed up
(2) Backup heaters will be deenergized

Proposed Answer: A

Explanation (Optional):

A. CORRECT.

B. Incorrect. Plausible because the Backup heaters will be energized, but the "A" charging pump will slow down

C. Incorrect. Plausible because the "A" charging pump will slow down, but the backup heaters will be energized.

D. Incorrect. Plausible if the candidate believes Average Tavg failed high.

Technical Reference(s): LP RIC01C AND RIC 03C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RIC01C 1.06 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 015 A4.02 | |
| | Importance Rating | 3.9 | |

Ability to manually operate and/or monitor in the control room: NIS indicators

Proposed Question: RO Question # 30

Plant conditions:

- The reactor tripped from 100% power due to a reactor coolant pump breaker failure.
- The crew is performing ES-0.1, Reactor Trip Response.
- Intermediate Range Nuclear Instrumentation Channel N-35 has stabilized at 8 $E^{-9}$ amps
- Intermediate Range Nuclear Instrumentation Channel N-36 is approaching 2 $E^{-10}$ amps

Which of the following describes how the Source Range Nuclear Instrumentation Channels (N31 and N32) will reinstate?

A. N31 and N32 must be manually reinstated by the HCO.

B. N31 and N32 will automatically reinstate on interlock from N-35.

C. N31 and N32 will automatically reinstate on interlock from P-10.

D. N31 will automatically reinstate on interlock from N35; N32 must be manually reinstated by the HCO.

Proposed Answer: A

Explanation (Optional):

A. CORRECT. Automatic reinstatement requires both IRNIS channels to be responding properly (less than 5 E-11 amps) since the logic for automatic reinstatement is 2/2.

B. Plausible if applicant believes the reinstatement logic is 1/2; the common logic for 2 channel system trips.

C. Plausible if applicant does not understand or properly apply the unblock reinstatement permissive provided by P-10. P-10 only allows manual or automatic reinstatement of

the SRNIS.

D. Plausible if applicant believes the reinstatement is via "channel matching" (N31/N35 and N32/N36).

Technical Reference(s): R3301C, Page 17 of 86 (Item 5) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3301C, 1.07 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Higher order because the applicant must apply the off-normal trip response to NIS/RPS design and determine that manual action is required to ensure proper NIS indication.

Meets K/A by requiring knowledge that manual action is required to ensure proper NIS indication following a plant trip.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 016 A3.01 | |
| | Importance Rating | 2.9 | |

Ability to monitor automatic operation of the NNIS, including: Automatic selection of NNIS inputs to control systems

Proposed Question: RO Question # 31

Plant conditions:

- 100% power
- Multiple alarms have actuated
- The plant is in a stable condition
- Turbine First Stage Pressure Channel PT-485 is reading 635 PSIG
- Turbine First Stage Pressure Channel PT-486 is reading 200 PSIG

Both feedwater control valves remain in AUTO with _____ providing the signal to the steam generator level control setpoint calculator.

A. PT-485

B. Average STEAM FLOW

C. Total FEEDWATER FLOW

D. Median select STEAM PRESSURE

Proposed Answer: A

Explanation (Optional):

A. CORRECT. PT-485 is the first choice as the ARBITRATE SELECT SIGNAL and within the acceptable range for the stated power.

B. Plausible because Average Steam Flow provides the ARBITRATE SIGNAL for 1st Stage Pressure and is the second choice as the ARBITRATE SELECT SIGNAL.

C. Plausible because Total Feedwater Flow provides the power signal for the selection between the HIGH and LOW Power mode of operation in ADFWCS.

D. Plausible because Avg Steam Pressure is used as an input signal in the ADFWCS and, much like 1st Stage Pressure, varies predictably from 0-100% power.

Technical Reference(s): LP R4401C Pg. 35 (6) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP R4401C 1.07.a.2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Meets K/A by requiring knowledge of the automatic selection of an NNIS input to a control system when an instrument failure occurs.

Identified as MEMORY level but borderline comprehension since applicant must know PT-485 is within the validation range to answer the question correctly.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 001 K2.05 | |
| | Importance Rating | 3.1 | |

Knowledge of bus power supplies to the following: M/G sets.....

Proposed Question: RO Question # 32

Plant conditions:

- 100% power with all components in a normal alignment
- A tagging operation is underway for a component fed by Bus 13

Which ONE of the following describes plant response if the AO performing the tagging mistakenly opens the Rod Drive MG Set Supply Breaker on Bus 13?

A. MG Set "A" trips and the reactor trips.

B. MG Set "A" trips and the plant continues to operate at 100% power.

C. MG Set "B" trips and the reactor trips.

D. MG Set "B" trips and the plant continues to operate at 100% power.

Proposed Answer: B

Explanation (Optional):

A. Plausible because the first part is correct; Bus 13 feeds MG "A" supply breaker. However, the MG Sets are in parallel feeding a series circuit so no trip occurs.

B. CORRECT. Bus 13 feeds MG "A" supply breaker. With MG "B" operating in parallel and feeding a series circuit no trip will occur.

C. Plausible if applicant does not know which bus (13 or 15) feeds which MG Set and does not apply the parallel circuit feeding a series circuit design.

D. Plausible if applicant does not know which bus (13 or 15) feeds which MG Set. The second part is correct.

Technical Reference(s): LP 3001C, Page 21/65, Item 2.a (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3001C, 1.05 (As available)

Question Source: Bank #

Modified Bank # 45562 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Modified to match GINNA plant design and changed format to four different combinations of two consequences.

Memory because of power supply recall and could answer by applying recall of E-0, REACTOR TRIP OR SAFETY INJECTION, immediate actions (RNO).

Meets K/A by requiring knowledge of power supply to one Rod Drive MG set and, for operational validity, how the circuit is designed.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 002 K1.07 | |
| | Importance Rating | 3.5 | |

Knowledge of the physical connections and/or cause-effect relationships between the RCS and the following systems: Reactor vessel level indication system

Proposed Question: RO Question # 33

Tcold provides density compensation and measurement of specific gravity for RCS fluid for the Reactor Vessel Level Indicating System (RVLIS) ...

A. during all modes of operation.

B. ONLY for operation with RCPs off.

C. ONLY for operation with NO SI or RHR Flow, and when CETs are NOT > Tsat.

D. ONLY for operation with RCPs running with NO SI or RHR Flow, and when CETs are NOT > Tsat.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Defeated with RCPs on or off if SI or RHR flow exists

B. Incorrect. Active for RCPs on or off

C. CORRECT.

D. Incorrect. Active also for RCPs off

Technical Reference(s): LP R6701C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R6701C, 1.07 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 071 K4.01 | |
| | Importance Rating | 2.6 | |

Knowledge of design feature(s) and/or interlock(s) which provide for the following: Pressure capability of the waste gas decay tank

Proposed Question: RO Question # 34

Which one of the choices identifies the pressure protection component installed on each Waste Gas Decay Tank discharge line that has the LOWER setpoint and where it discharges to?

A. Relief Valve; Plant Vent

B. Relief Valve; Vent Header

C. Rupture Disk; Plant Vent

D. Rupture Disk; Vent Header

Proposed Answer: D

Explanation (Optional):

A. Plausible as one of two pressure protection components for each Waste Gas Decay Tank and it does relieve to the Plant Vent. However, at 150 PSIG it is set higher than the Rupture Disk (142 PSIG).

B. Plausible as one of two pressure protection components for each Waste Gas Decay Tank. However, at 150 PSIG it is set higher than the Rupture Disk (142 PSIG) and it is the Rupture Disk that relieves to the Vent Header.

C. Plausible because the first part is correct: Rupture Disk @ 142 PSIG and Relief Valve @ 150 PSIG. However, the relief valve lifts to the Plant Vent not the Rupture Disk.

D. CORRECT. The Rupture Disk fails @ 142 PSIG and Relief Valve lifts @ 150 PSIG. The Rupture Disk relieves pressure back to the Vent Header. The Relief Valve relieves pressure back to the Plant Vent.

Technical Reference(s): LP 3801C, Pages 22 (bottom) and (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3801C (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Recall of relative value of setpoints. Stayed away from recall of setpoints since all system operations are performed in the field or at a local panel, by AO's.

Meets K/A by requiring knowledge of pressure protection component setpoints and their flowpaths.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 072 K3.02 | |
| | Importance Rating | 3.1 | |

Knowledge of the effect that a loss or malfunction of the ARM system will have on the following: Fuel handling operations

Proposed Question: RO Question # 35

Plant conditions:

- A refueling outage has started
- Core off-load is in progress
- E-24, RMS Area Monitor High Activity, has actuated
- Operators have determined that the FAIL ALARM has actuated on Radiation Monitor R-2, Containment Area Monitor
- Parts to repair R-2 are estimated to arrive on site in approximately 3 days

With R-2 failed, core off-load ...

A. must be stopped until R-2 is restored to operability.

B. must be stopped but may resume when a local monitor is installed on the manipulator bridge.

C. may continue if R-7, In-Core Detectors Area Monitor, is operable.

D. may continue if either R-29 or R-30, Containment High Range Monitor, is operable.

Proposed Answer: B

Explanation (Optional):

A. Plausible because it is partially correct. However, the procedure allows a local monitor on the manipulator bridge to be substituted.

B. CORRECT. O-15.1 specifies either R-2 or a local monitor on the manipulator bridge.

C. Plausible because a substitute monitor is permitted. R-7 is a monitor inside containment but is NOT the designated substitute.

D. Plausible because a substitute monitor is permitted. R-29 and R-30 are monitors inside containment but are NOT the designated substitute.

Technical Reference(s): O-15.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3701C, 1.09 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12

55.43

Radiological safety principles and procedures.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 041 A2.02 | |
| | Importance Rating | 3.6 | |

Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations: Steam valve stuck open

Proposed Question: RO Question # 36

Plant conditions:

- Reactor power is stable at 1% power while reactor engineers gather data during a reactor startup
- One previously closed condenser steam dump valve has failed full open
- SG pressure is reduced to 965 psig prior to reclosing the valve
- RCS temperature indicates 541.0°F on Loop 'A' and 541.5°F on Loop 'B'

RCS temperature is ___(1)___.
RCS temperature will be restored by ___(2)___.

A. (1) above the technical specification minimum temperature for criticality
(2) condenser steam dump valve closure and/or manual rod withdrawal

B. (1) above the technical specification minimum temperature for criticality
(2) automatic rod withdrawal

C. (1) below the technical specification minimum temperature for criticality
(2) condenser steam dump valve closure and/or manual rod withdrawal

D. (1) below the technical specification minimum temperature for criticality
(2) automatic rod withdrawal

Proposed Answer: A

Explanation (Optional):

A. CORRECT. Rod Control is in MANUAL at less than 12.8% power. Also, Minimum temperature for criticality is 540 degrees F.

B. Incorrect. Plausible because the first part is correct. However, automatic rod control is not available until reactor power is above 12.8%.

C. Incorrect. Plausible because the action is correct but the temperature is incorrect. Applicant can confuse temperatures because no load is 547 and steam dumps close on low temperature with reactor trip breakers open to ensure inadvertent cooldown does not occur.

D. Incorrect. Plausible as described in Options above.

Technical Reference(s): O-1.2
TS 3.4.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R4501C, 1.06 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Meets K/A by requiring knowledge of how the rest of the steam dump system and the integrated plant responds to a steam dump valve failure at low power.

Higher order because the applicant must apply heat transfer, reactivity balance and steam dump system design concepts to answer the question.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 055 K1.06 | |
| | Importance Rating | 2.6 | |

Knowledge of the physical connections and/or cause effect relationships between the CARS and the following systems: PRM system

Proposed Question: RO Question # 37

Which ONE of the following radiation monitors' alarm setpoints are set to ensure that Technical Specification limits for steam generator tube leakage are not exceeded?

A. R-31/R-32, Main Steam Line

B. R-19, Steam Generator Blowdown

C. R-47, Air Ejector/Exhaust

D. R-48, Air Ejector/Exhaust

Proposed Answer: C

Explanation (Optional):

A. Incorrect. R31 and R32 are good indicators of which SG is ruptured as a backup to other indications of an SGTR

B. Incorrect. R19 activity is helpful in determining trend of leak rate but is not used or calibrated for TS leakage limits

C. Correct.

D. Incorrect. R48 is used as a high range detector that is helpful in establishing EAL classifications

Technical Reference(s): R3901C
P-9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R4301C, 1.06.b
R3901C, 1.08.a (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 11
55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Also meets 10CFR55.41(b) item 4

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 086 A1.04 | |
| | Importance Rating | 2.7 | |

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Fire Protection System operating the controls including: Fire dampers

Proposed Question: RO Question # 38

Plant conditions:

- The unit is in Mode 5, heating up to Mode 4
- Fire Damper BB-46-P in Battery Room B has just been declared INOPERABLE

Which ONE of the following meets Technical Specification/TRM requirements for this condition?

Fire Damper BB-46-P ...

A. must be placed in the closed position within one hour and then "B" Battery must be declared INOPERABLE.

B. must be placed in the closed position within one hour. The unit must not enter Mode 4 until all fire barrier penetration seals are OPERABLE.

C. can remain open but a continuous fire watch must be established on one side of the damper within one hour.

D. can remain open. Verify all Battery Room "B" fire detectors OPERABLE within one hour then perform a once-per-shift fire watch inspection of "B" Battery Room.

Proposed Answer: C

Explanation (Optional):

A. Plausible because it is typical for failed components to be placed in the safety-response position. In this case, if the damper is closed then Battery Ventilation and therefore operability must be evaluated. However, the damper is NOT required to be closed.

B. Plausible because it is typical for failed components to be placed in the safety-response position and, generally, mode changes are not permitted with reliance on action statements. However, fire barrier penetration seals are required to be operable in all

modes (not just Mode 4 and higher) and the damper is NOT required to be closed.

C. CORRECT. The damper is NOT required to be placed in the safety response position. One hour action per TRM TR 3.7.5, Condition A.

D. Plausible because the first part is correct and the second part is partially correct. However, a once-per-hour fire watch inspection is required.

Technical Reference(s): TRM TR 3.7.5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP R5901C, 1.12 (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Meets K/A indirectly. Knowledge of compensatory action(s) required because the position of the fire damper will NOT be changed in order that battery ventilation be maintained. Licensed operators have limited responsibility for recall of specific fire damper operations but are required to recall one hour actions from TS/TRM.

Bank Question (attached) used as a basis for this question in order to be "in the ballpark" of expected operator knowledge for this system.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 009 EK1.02 | |
| | Importance Rating | 3.5 | |

Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Use of steam tables

Proposed Question: RO Question # 39

Plant conditions:

- A LOCA has occurred.
- A manual Rx trip and Safety Injection were initiated
- RCS pressure is 1085 psig.
- CETs indicate 525°F.
- Containment pressure is 7 psig.
- Transition to ES-1.1, SI Termination, is being evaluated by the CRS.

Which ONE of the following identifies current RCS subcooling in accordance with Figure 1.0, and whether all conditions are met to transition to ES-1.1?

(Reference Provided)

A. RCS subcooling is 0°F; transition will be made

B. RCS subcooling is 10°F; transition will be made

C. RCS subcooling is 0°F; transition will NOT be made

D. RCS subcooling is 10°F; transition will NOT be made

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Subcooling will indicate higher than zero. The applicant may be off by a gradient on the figure. Transition will <u>not</u> be made with zero subcooling, but transition is plausible if applicant doesn't consider that greater than zero required.

B. Incorrect. Plausible because if applicant uses normal or adverse containment values, they will believe transition may be made.

C. Incorrect. Subcooling will indicate higher than zero. The applicant may be off by a gradient on the figure. Plausibility enhanced because if the applicant arrives at zero and correctly applies the transition, this option would be chosen

D. Correct. Plausible because the value is correct, but if value is greater than zero, the transition would be made based on subcooling alone, but will NOT be made under this condition where RCS pressure is too low.

Technical Reference(s): E-1 Figure 1.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Figure 1.0

Learning Objective: REP00C, Obj 2.01 (As available)

Question Source: Bank # WTSI 64256

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Salem 2006

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 007 EK1.04 | |
| | Importance Rating | 3.6 | |

Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Decrease in reactor power following reactor trip (prompt drop and subsequent decay)

Proposed Question: RO Question # 40

Which ONE of the following describes Nuclear Instrumentation response to a reactor trip from 100% power until the Source Range instruments energize, from the time control rods begin to drop?

A. Prompt Drop of approximately 3 decades, followed by a -1/3 DPM startup rate for approximately 20 minutes.

B. Prompt Drop of approximately 3 decades, followed by a -1/3 DPM startup rate for approximately 3-4 hours.

C. Prompt Drop to approximately 5% power, followed by a -1/3 DPM startup rate for approximately 3-4 hours.

D. Prompt Drop to approximately 5% power, followed by a -1/3 DPM startup rate for approximately 20 minutes.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because the applicant may confuse the magnitude of the prompt drop (3 decades would make power approximately 0.1%, which nearly 0%, similar to 5% being near 0%) Additionally, the -1/3 DPM SUR for 20 minutes is correct.

B. Incorrect. Plausible same reason as option A, and also because power does show a noticeable decrease for 3-4 hours, but the rate will be lower after the first 20 minutes.

C. Incorrect. Plausible because the prompt drop is correct, and also for same reason as option B for time that SUR is -1/3 DPM.

D. Correct. Decay heat is approximately 5-7%, and when the reactor trips, that will be the immediate indication. Short Lived delayed neutron precursors will cause the Startup Rate to be -1/3 DPM for approximately 20 minutes until they have decayed sufficiently,

the SUR lowers for about 3-4 hours until power is stable in Source Range.

Technical Reference(s): RRT04C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

Question Source: Bank # WTSI 52812

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Harris 2004 NRC

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 1

55.43

Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | E05 EK1.1 | |
| | Importance Rating | 3.8 | |

Knowledge of the operational implications of the following concepts as they apply to the (Loss of Secondary Heat Sink) Components, capacity, and function of emergency systems.

Proposed Question: RO Question # 41

Plant conditions:

- A loss of Feedwater has resulted in a reactor trip.
- The crew is performing actions of FR-H.1, Loss of Secondary Heat Sink.

Which ONE of the following describes the operation of the PRZR PORVs during this event? Pressurizer PORVs ...

A. may be operated to depressurize the RCS so that Low Steam Pressure and Low PRZR Pressure SI signals can be blocked; this prevents unwanted SI actuation when depressurizing SGs prior to performing action to establish Main Feedwater flow.

B. may be operated to depressurize the RCS so that Low Steam Pressure and Low PRZR Pressure SI signals can be blocked; this prevents unwanted SI actuation when depressurizing SGs prior to performing action to establish Condensate flow.

C. remain closed with block valves closed to preserve RCS inventory unless conditions exist that require initiation of Bleed and Feed cooling of the RCS.

D. remain in automatic and allowed to operate with block valves open for PRZR pressure control. One PORV is opened if Bleed and Feed cooling is required, with the other PORV remaining available for automatic control.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the phrasing is similar, but no depressurization is required in order to initiate Feedwater flow

B. Correct. Aux Spray is the preferred method here, but one PORV may be used if Aux Spray is not available (Step 10)

C. Incorrect. Plausible because opening a PORV can possible lead to loss of inventory. The only way block valves will be closed is if a PORV is failed or leaking.

D. Incorrect. Plausible because this is similar to the use of PORVs in FR-C.1 and FR-C.2. The second part of this option is correct.

Technical Reference(s): FR-H.1, Rev 38, Step 10 RNO (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RFRH1C, Obj 1.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | E11 EK2.2 | |
| | Importance Rating | 3.9 | |

Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, and the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question: RO Question # 42

Which ONE of the following describes the reason and requirements for the INITIAL depressurization of the RCS per ECA-1.1, Loss of Emergency Coolant Recirculation?

The initial depressurization is done to ___(1)_____.
The crew will ____(2)____.

A. (1) ensure SI Accumulator injection.
(2) depressurize the RCS maintaining minimum RCS subcooling, then terminate SI pump flow.

B. (1) minimize RCS leakage.
(2) depressurize the RCS maintaining minimum RCS subcooling, then reduce SI pump flow.

C. (1) minimize RCS leakage.
(2) depressurize the RCS maintaining maximum RCS subcooling, then stabilize RCS temperature while attempting to restore makeup sources.

D. (1) ensure SI Accumulator injection.
(2) depressurize the RCS maintaining maximum RCS subcooling, then stabilize RCS temperature while attempting to restore makeup sources.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because setup for accumulator injection is performed later in procedure, and depressurizing will ensure injection at some point, but SI is not secured, only minimized

B. Correct. The depressurization is performed to decrease leakage, therefore decreasing RCS makeup requirements.

C. Incorrect. Maximizing subcooling will delay depressurization, and will not stabilize RCS temperature. Plausible because normally RCS subcooling is maximized

D. Incorrect. Plausible because setup for accumulator injection is performed later in the procedure after SG depressurization, but the crew will not stabilize RCS temperature.

Technical Reference(s): ECA-1.1, ECA-1.1 Background Doc (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: REC11C, Obj 1.01 (As available)

Question Source: Bank # WTSI 64936

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Wolf Creek 2009 NRC

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | E04 EK2.1 | |
| | Importance Rating | 3.5 | |

Knowledge of the interrelations between the (LOCA Outside Containment) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO Question # 43

Plant conditions:

- The plant was initially at 100% power.
- A Reactor Trip and Safety Injection have occurred.
- RCS pressure is 1200 psig.
- R-14, Plant Effluent Gas Monitor, is in alarm.
- Area Radiation Monitor Readings in the Auxiliary Building are increasing.
- The Crew has transitioned from E-0 to ECA-1.2, LOCA Outside Containment.

Why does ECA-1.2 direct closing RHR Discharge to Reactor Vessel Deluge valve, MOV-852A?

Closing MOV-852A will ...

A. assist in establishing a Hot Leg Injection flow path

B. determine if the leak is in the RHR discharge piping

C. prevent RWST inventory from being lost to the environment if the break is in the RHR discharge piping

D. maintain RHR discharge piping isolated when RHR pumps are stopped because they are not required due to RCS pressure

Proposed Answer: B

Explanation (Optional):

A. Incorrect. ECA-1.2 does not align hot leg injection. MOV-852A would be closed for hot leg injection but not in this procedure. Plausible because it isolates cold leg injection from RHR

B. Correct. Purpose is to locate and isolate the leak

C. Incorrect. Plausible because there is a leak and RHR takes suction from RWST, but in this condition, RCS is leaking, not RWST.

D. Incorrect. RHR would only remain isolated if closing the valve stops the leak. Plausible because RCS pressure is above RHR pump shutoff head, and they will be shut down in E-1, but the valve is not closed due to securing RHR pumps

Technical Reference(s): ECA-1.2, Step 3, Background (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: REC12C, Obj 1.02 (As available)

Question Source: Bank #

Modified Bank # WTSI 19024 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Robinson 2002 NRC, modified by adding condition to stem so that distracter options could be made more plausible

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 077 AK2.07 | |
| | Importance Rating | 3.6 | |

Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine / generator control

Proposed Question: RO Question # 44

Plant conditions:

- The plant is operating at 100% power.
- Severe weather is in the area.
- Grid instabilities are being reported.
- Circuit 911 is lost due to a lightning strike.
- RG&E ECC is evaluating conditions for potential power restrictions on the grid.

Which ONE of the following describes the action that will be required in accordance with O-6.9, Ginna Station Operating Limits for 13A Transmission?

A. Refer to the applicable attachment to determine operability of off-site power sources and adjust Main Generator Excitation if required.

B. Start and load both EDGs due to inoperability of 480 volt Safeguards Buses.

C. Reduce Main Generator Output in accordance with RG&E instructions if ECC determines that net generation is too high.

D. Trip the reactor; enter E-0, Reactor Trip or Safety Injection.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because this would be performed if a Low Voltage Contingency alarm was received, and is covered by the same procedure

B. Incorrect. Plausible because this would be performed if it was determined that off-site power was inoperable and affected the safeguards bus operability.

C. Correct. RG&E ECC may call to require a load reduction when an off-site line is lost

because net generation may be too high for degraded grid capacity

D. Incorrect. The reactor would only be tripped if the facility cannot reduce net generation within the time period committed to.

Technical Reference(s): O-6.9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R4101C, 1.11 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 022 AK3.03 | |
| | Importance Rating | 3.1 | |

Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Pump Makeup: Performance of lineup to establish excess letdown after determining need

Proposed Question: RO Question # 45

Plant conditions:

- The plant is at 100% power when a small RCS leak develops.
- The crew takes the necessary actions to restore and stabilize PRZR level.
- Subsequently, the crew attempts to determine the leakage source in accordance with the appropriate AP.
- After isolating letdown, the crew observes the following parameters/indications:
    - Letdown flow 0 gpm
    - Containment pressure approximately 0.8 psig and lowering
    - RCS pressure approximately 2235 psig and stable
    - PRZR level approximately 72% and rising
    - Charging flow 40 gpm
    - Containment radiation monitors trending downward slowly

Which ONE of the following describes the required actions in accordance with AP-CVCS.1, CVCS Leak?

A. Immediately trip the reactor and enter E-0, Reactor Trip or Safety Injection because PRZR level exceeds the maximum limit.

B. Place Excess Letdown in service and adjust seal injection flow to maintain PRZR level within limits.

C. Initiate Shutdown per O-2.1, Normal Shutdown to Hot Shutdown, due to RCS leakage exceeding limits.

D. Re-establish normal Letdown and dispatch leak search team to identify leakage source.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Applicant may mistake 72% PRZR level for the reactor trip value; however, the initial conditions do not require an immediate reactor trip.

B. Correct. The RCS leak was determined to be on normal letdown based on containment pressure decreasing and radiation monitor readings stabilized. Therefore to maintain pressurizer level and RCS chemistry (chemical feed and boration), excess letdown is required.

The applicant must understand from the initial conditions the RCS leak was on the letdown line and has been isolated. In order to maintain pressurizer level and chemistry excess letdown is required.

C. Incorrect. Since pressurizer level is stabilized and is controllable by placing Excess Letdown in service, the other initial conditions indicated the RCS leak has been isolated a shutdown is not required.

D. Incorrect. The leak has been isolated to the normal letdown line. Therefore, a leak search team is unnecessary.

Technical Reference(s): AP-CVCS.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP05C, Obj 2.01 (As available)

Question Source: Bank # WTSI 64349

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Sequoyah 2008

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 026 AK3.03 | |
| | Importance Rating | 4.0 | |

Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Guidance actions contained in EOP for Loss of CCW/nuclear service water

Proposed Question: RO Question # 46

Plant conditions:

- The plant is in Mode 4.
- A loss of CCW has occurred.
- The crew is performing AP-CCW.3, Loss of CCW- Plant Shutdown.
- The HCO is ensuring that seal injection flow is maintained to RCPs.
- A cooldown is being initiated as the crew attempts to restore CCW.

Which ONE of the following describes the requirement for maintaining seal injection in this procedure, and the reason for the requirement?

Maintain seal injection to RCPs until RCS temperature reaches...

A. 150°F; To ensure a high pressure source of water to the seal area to keep CRUD from the RCS from fouling the seal package.

B. 150°F; To ensure RCP seal cooling is maintained.

C. 250°F; To ensure a high pressure source of water to the seal area to keep CRUD from the RCS from fouling the seal package.

D. 250°F; To ensure RCP seal cooling is maintained.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. This is true for an idle RCP in cold shutdown or hot standby but for a loss of CCW, the concern for fouling a seal is subordinate to seal cooling. Plausible because it is an RCP action and the temperature is correct.

B. Correct. Procedure recommends ensuring seal injection is maintained until RCS is

below 150.

C. Incorrect. This is true for an idle RCP in cold shutdown or hot standby but for a loss of CCW, the concern for fouling a seal is subordinate to seal cooling. Plausible because it is an RCP action.

D. Incorrect. Westinghouse recommends that some seal cooling be maintained above 150°F. If seal injection was terminated at 250, then the RCP could be damaged.

Technical Reference(s): AP-CCW.2 Note and Background Information (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP03C 1.03 (As available)

Question Source: Bank # WTSI 66509

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Ginna 2006

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 040 AK3.02 | |
| | Importance Rating | 4.4 | |

Knowledge of the reasons for the following responses as they apply to the Steam Line Rupture: ESFAS initiation

Proposed Question: RO Question # 47

Which ONE of the following describes the reason for automatic Safety Injection actuation during a Main Steam Line Break (MSLB)?

A. Maintains RCS pressure to prevent loss of subcooling and reactor vessel head voiding during the RCS cooldown.

B. Minimizes the probability of a Pressurized Thermal Shock event by limiting the pressure transient on the RCS.

C. Ensures that a Containment Isolation signal will be generated in the event of a MSLB inside Containment.

D. Ensures borated water is added to the RCS to offset the positive reactivity added during the RCS cooldown.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because head voiding is a concern during the performance of some EOPs. In this case, subcooling will be high as the RCS cools down and the RCS will shrink. But, because it a MSLB, inventory is not lost and head voiding is unlikely.

B. Incorrect. SI increases chance of PTS because once the faulted SG is fully depressurized, pressure will be high, and heatup will squeeze the RCS causing pressure to rise further with SI inventory already in the RCS

C. Incorrect. Plausible because it is a true statement by itself, but is not the reason that Safety Injection is actuated during a Main Steam Line Break

D. Correct. SI actuates to ensure boron counteracts the positive reactivity from a severe cooldown of the RCS

Technical Reference(s): ITS 3.5.2 Basis (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R2701C 1.13 (As available)

Question Source: Bank # WTSI 19018

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Robinson 2002

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Modified distracters for plausibility

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 056 AA1.24 | |
| | Importance Rating | 2.9 | |

Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: Plant computer, to call up in-core temperature monitoring group

Proposed Question: RO Question # 48

Plant conditions:

- Reactor trip followed by a loss of off-site power.
- The crew is performing the actions of ES-0.2, Natural Circulation Cooldown.
- The HCO is monitoring the RCS cooldown.

Which ONE of the following describes the use of the PPCS to monitor the RCS cooldown in accordance with A-503.1, Emergency and Abnormal Operating Procedures User's Guide?

A. PPCS is the required means of plotting the cooldown using the Core Exit Thermocouples page

B. PPCS is the primary means of plotting the cooldown using Th and Tcold trending using the RCS display

C. MCB indication is the primary source of information during EOP use, but PPCS may be used to monitor Core Exit Thermocouples to monitor RCS subcooling as desired

D. MCB indication is the primary source of information during EOP use. Information provided by PPCS may NOT be acted upon UNLESS MCB indicators are consistent with that information

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Not required, only used as backup indication, but PPCS may be used, it is just not required to be used

B. Incorrect. RCS display doesn't trend, and PPCS not primary indicator in accordance with A503.1

C. Correct.

D. Incorrect. There is SPDS info that can only be gotten from PPCS, and is acted upon during performance of EOPs

Technical Reference(s): A-503.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3201C, Obj 1.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 015 AA1.21 | |
| | Importance Rating | 4.4 | |

Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Development of natural circulation flow

Proposed Question: RO Question # 49

Plant conditions:

The plant was operating at 100% power when a reactor trip and subsequent loss of off-site power occurred.

Which ONE of the following describes the effect on RCS temperature immediately after the RCPs trip as natural circulation is developing?

A. Thot rises until Natural Circulation develops, then lowers as decay heat load lowers. Loop Delta T remains constant throughout the event.

B. Tcold lowers until Natural Circulation develops, then stabilizes as decay heat load lowers. Loop Delta T remains constant throughout the event.

C. Thot rises until Natural Circulation develops, then lowers as decay heat load lowers. Loop Delta T lowers as decay heat load lowers.

D. Tcold lowers until Natural Circulation develops, then stabilizes as decay heat load lowers. Loop Delta T lowers as decay heat load lowers.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. First part is correct. Second part is plausible if the applicant misunderstands Tcold behavior. Tcold will track SG pressure

B. Incorrect. Tcold actually rises immediately as the SG pressure rises due to Turbine stop valves closing and reduced steam demand against the initial reactor decay heat. Description of Loop Delta T same as in option A

C. Correct. When the reactor trips and RCPs are lost, there is less flow through the core to remove decay heat. As natural circulation sets up, Thot becomes hotter because the

water spends more time in the core with a low flow rate. Natural circulation develops due to density difference between Thot and Tcold, and as decay heat lowers and natural circulation flow rises, Thot lowers because there is less decay heat generated and the water spends less time in the core.

D. Incorrect. Same description as Option B, except that loop Delta T description is correct

Technical Reference(s): Westinghouse Executive Volume - Natural Circulation (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 14

55.43

Principles of heat transfer, thermodynamics and fluid mechanics.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 055 EK3.02 | |
| | Importance Rating | 4.3 | |

Knowledge of the reasons for the following responses as the apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power

Proposed Question: RO Question # 50

Plant conditions:

Which ONE (1) of the following is a purpose of depressurizing all intact SGs to 260 psig during the performance of ECA-0.0, Loss of All AC Power?

A. Reduces DP across SG U-tubes to minimize RCS inventory loss due to tube rupture.

B. Reduces DP across RCP seals to minimize leakage and loss of RCS inventory

C. Maximizes Natural Circulation flow before Reflux cooling begins as the RCS becomes saturated.

D. Maximizes Natural Circulation flow to allow reactor vessel head to cool since CRDM cooling fans are unavailable.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. The most likely failure for this event is loss of inventory through failed RCP seals not SGTR

B. Correct.

C. Incorrect - steaming is a method to increase natural circ and would occur, however minimizing inventory loss is a greater concern at this point.

D. Incorrect - steaming is a method to increase natural circ and would occur, however minimizing inventory loss is the greater concern at this point.

Technical Reference(s): ECA-0.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: REC00C, Obj 2.01 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 057 AA2.17 | |
| | Importance Rating | 3.1 | |

Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: System and component status, using local or remote controls

Proposed Question: RO Question # 51

Plant conditions:

- The plant is operating at 100% power.
- The following alarms are received in the control room:
  - E-3, INVERTER TROUBLE
  - E-6, LOSS A INSTR BUS
- Instrument Bus 'A' voltage is zero volts.

Which ONE of the following describes the effect on the plant?

A. All RED (Channel 1) bistable lights on the Status Panel are LIT; Rod Control remains available in AUTO

B. All RED (Channel 1) bistable lights on the Status Panel are LIT; Rod Control must be placed in MANUAL

C. All RED (Channel 1) and WHITE (Channel 2) bistable lights on the Status Panel are OFF; Rod Control remains available in AUTO

D. All RED (Channel 1) and WHITE (Channel 2) bistable lights on the Status Panel are OFF; Rod Control must be placed in MANUAL

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Rod Control must be placed in manual because the +/- 4°F auto rod stop will be actuated. One set of indications supplied by Instrument Bus A will be lost.

B. Correct. See P-12, section 2.5

C. Incorrect. Indication is for a loss of Instrument Bus B, which supplies power to the bistable lights. Also, manual rod control is required as described in Option A

D. Incorrect. Indication is for a loss of Instrument Bus B, which supplies power to the bistable lights.

Technical Reference(s): P-12
ER-INST.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R0901C, Obj 1.11 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 027 AA2.12 | |
| | Importance Rating | 3.7 | |

Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR level

Proposed Question: RO Question # 52

Plant conditions:

- The plant is operating at 100% power.
- The following alarm is received in the control room:
  - F-19, PRZR PORV OUTLET HI TEMP 145°
- The HCO determines that PORV PCV-431C indicates open.
- PRZR pressure indicates 2000 psig and lowering.
- The HCO trips the reactor.
- The PORV is isolated when PRZR pressure has lowered to 825 psig.
- Safety Injection failed to automatically actuate, and was manually actuated at the same time that the PORV was closed.

Which ONE of the following describes indicated PRZR level at the time the PORV is isolated, and whether PRZR level is an accurate indication of RCS inventory?

A. Off-Scale low or lowering rapidly; if level is on scale, it is considered an accurate indication of RCS inventory.

B. Off-Scale low or lowering rapidly; if level is on scale, it is NOT considered an accurate indication of RCS inventory.

C. Off-Scale high or rising rapidly; if level is on scale, it is considered an accurate indication of RCS inventory.

D. Off-Scale high or rising rapidly; if level on scale, it is NOT considered an accurate indication of RCS inventory.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Pressure has decreased to a value below saturation pressure. Therefore, with RCPs tripped, a bubble will form under the head. Water will be expelled through the pressurizer opening, resulting in high indicated level. Water level only lowers prior to reaching saturation in the RCS.

B. Incorrect. Same discussion as in option A, but second half of the statement is correct...

C. Incorrect. Level indication is correct, but it will not be an accurate indication, particularly if SI was not actuated when required.

D. Correct. Saturation has been reached under the vessel head. PRZR level will rise as a bubble forms under the head. It is not an accurate indication of RCS inventory because RVLIS will show that there is voiding. If pressure was raised, PRZR level would fall

Technical Reference(s): UFSAR 15.6.1.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 038 EA2.01 | |
| | Importance Rating | 4.1 | |

Ability to determine or interpret the following as they apply to a SGTR: When to isolate one or more S/Gs

Proposed Question: RO Question # 53

Plant conditions:

- A Steam Generator Tube Rupture is in progress.
- The crew is performing actions of E-3, Steam Generator Tube Rupture.
- SG 'A' has been identified as ruptured.
- SG A Narrow Range level - 8%
- SG B Narrow Range level - 3%

In accordance with E-3, which ONE of the following describes the EARLIEST time that AFW flow may be isolated to SG 'A'?

A. Immediately.

B. ONLY when both SG Narrow Range levels are above 7%.

C. Not until 'A' SG narrow range level is above 25%.

D. ONLY when both SG narrow range levels are above 25%.

Proposed Answer: A

Explanation (Optional):

A. Correct. ≥7% in the ruptured SG is required prior to isolation of that SG, to ensure SG tubes are covered and a thermal layer may be established above the break.

B. Incorrect. The ruptured SG level must be ≥ 7% to ensure an adequate thermal layer exists prior to isolation of Aux Feed. This will insulate the steam space in the SG from the cooler RCS water as the RCS is cooled down to allow depressurization of the RCS to stop the leakage to the SG without losing subcooling, if the SG were to depressurize while trying to equalize RCS to SG pressure. 7% is NOT required in a steam generator

that is unaffected.

C. Incorrect. Plausible because this value is listed in the procedure as the value that would be used if adverse containment conditions existed.

D. Incorrect. Plausible because both SGs are desired fro symmetrical heat removal on a reactor trip, but to isolate a ruptured SG, only the affected SG must be ≥7%. The applicant would choose this if they believe that ≥25% in the unaffected SG is required to maintain minimum heat removal capability. 25% will be familiar, because it is the value required if containment conditions were adverse.

Technical Reference(s): E-3, Rev 46, Step 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: REP03C, Obj 2.01 (As available)

Question Source: Bank # WTSI 65819

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 011 2.1.23 | |
| | Importance Rating | 4.3 | |

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: RO Question # 54

Plant conditions:

- A LOCA has occurred.
- The crew is performing actions of E-0, Reactor Trip or Safety Injection.
- RCS pressure is 200 psig.
- The crew has just verified immediate actions.
- ECCS is operating as designed.
- Containment pressure is 31 psig and rising.
- Annunciator A-27, CNMT SPRAY, is LIT.
- BOTH Containment Spray Pumps GREEN breaker indicating lights are LIT.
- The HCO reports there is NO Containment Spray flow on either train.

Which ONE of the following describes (1) the MINIMUM action required to manually actuate Containment Spray, and (2) the action required in accordance with E-0 for operation of RCPs?

A. (1) Depress either 1 of 2 Containment Spray actuation pushbuttons
(2) Leave RCPs running until evaluating status at the appropriate step in E-0

B. (1) Depress either 1 of 2 Containment Spray actuation pushbuttons
(2) IMMEDIATELY trip both RCPs based on E-0 Foldout page requirements

C. (1) Depress BOTH Containment Spray actuation pushbuttons
(2) Leave RCPs running until evaluating status at the appropriate step in E-0

D. (1) Depress BOTH Containment Spray actuation pushbuttons
(2) IMMEDIATELY trip both RCPs based on E-0 Foldout page requirements

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Both pushbuttons must be depressed at the same time. This minimizes the

probability of an inadvertent spray actuation. Second part is incorrect.

B. Incorrect. Both pushbuttons must be depressed at the same time. This minimizes the probability of an inadvertent spray actuation. RCP operation is plausible because it is common to trip RCPs as part of spray actuation step due to loss of CCW.

C. Incorrect. RCP evaluation is a continuous action step.

D. Correct.

Technical Reference(s): E-0, Rev 43, Steps 5
E-0 Foldout Page (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: REP00C, Obj 2.01 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 054 2.4.34 | |
| | Importance Rating | 4.2 | |

Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: RO Question # 55

Plant conditions:

- The reactor was tripped due to loss of all CCW to 'B' RCP.
- CCW CANNOT be restored to 'B' RCP.
- Subsequently, a loss of Feedwater occurred.
- The crew is performing FR-H.1, Loss of Secondary Heat Sink.
- MDAFW Pumps are tripped and CANNOT be restarted.
- TDAFW Pump steam supply valves MOV-3504A and 3505A are OPEN.
- V-3652, Turb Drvn AFW Pump Governor Vlv, indicates GREEN light lit, RED light extinguished.
- TDAFW flow is 0 gpm
- TDAFW discharge pressure is 0 psig
- SG levels are 45 inches in BOTH SGs.
- Bleed and Feed has been initiated.
- RCS temperature is lowering slowly.

For the plant condition, which ONE of the following describes the local operator actions that are required in accordance with FR-H.1, and if the actions are unsuccessful, the NEXT action that will be required to restore secondary heat sink?

Locally reset TDAFW Pump governor valve...

A. ONLY; if AFW flow cannot be restored, attempt to initiate Main Feedwater flow

B. ONLY; if AFW flow cannot be restored, attempt to initiate Standby Aux Feedwater flow

C. AND close 'B' RCP seal injection needle valve; if AFW flow cannot be restored, attempt to initiate Main Feedwater flow

D. AND close 'B' RCP seal injection needle valve; if AFW flow cannot be restored, attempt to initiate Standby Aux Feedwater flow

Proposed Answer: ~~D~~ B correct, per revised key. J. Fink

Explanation (Optional):

A. Incorrect. Plausible because the governor valve will be reset and also because Main Feedwater flow is the next action required

B. Incorrect. Plausible because the governor valve will be reset. Standby AFW is not initiated at this point in the event. The next action after AFW is Main feedwater

C. Incorrect. Plausible because all actions are correct except initiation of Main Feedwater is not the next step.

D. Correct.

Technical Reference(s): FR-H.1, Rev 38 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RFRH1C, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 008 2.1.32 | |
| | Importance Rating | 3.8 | |

Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: RO Question # 56

Plant conditions:

- Reactor Trip and Safety Injection have actuated due to a LOCA.
- The crew is performing actions of ES-1.2, Post LOCA Cooldown and Depressurization.
- PRZR level is off-scale high.
- RCS subcooling indicates 5°F superheat.
- RCS cooldown is in progress.
- During the cooldown, an ORANGE condition is indicated on the INTEGRITY Critical Safety Function Status Tree.
- The CRS determines that the crew will remain in ES-1.2 until RHR injection is attained, before addressing the Orange condition.

Which ONE of the following describes the reason for the decision to remain in ES-1.2?

A. Since RCS subcooling is negative, Pressurized Thermal Shock is NOT an immediate concern, even if RCS cooldown rates are high.

B. Once RHR injection is attained, the Orange condition on the Integrity Critical Safety Function Status Tree will clear, and the actions of the FR procedure will not be required.

C. Challenges to the Integrity Critical Safety Function are expected when performing the actions of ES-1.2 for a pressurizer vapor space LOCA.

D. Once an operator controlled cooldown is underway, the likelihood of Pressurized Thermal Shock is reasonably remote.

Proposed Answer: A

Explanation (Optional):

A. Correct. Caution prior to step 7 says not to go to FR-P.1 until the cooldown is complete to the point of RHR injection

B. Incorrect. Plausible because it sounds reasonable that RHR injection will mean no PTS. However, the CSF Status Tree looks at cooldown rate and if the max rate is exceeded, it is possible to get an orange or red condition

C. Incorrect. RCS cooldown rates are expected to be below the rate required to receive a challenge to the Integrity CSFST

D. Incorrect. Plausible because it sounds similar to reason for RCP trip criteria not applying once a cooldown is started in E-3

Technical Reference(s): ES-1.2 Background, caution prior to step 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RES12C, Obj. 1.02 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 032 AK1.01 | |
| | Importance Rating | 2.5 | |

Knowledge of the operational implications of the following concepts as they apply to Loss of Source Range Nuclear Instrumentation: Effects of voltage changes on performance

Proposed Question: RO Question # 57

Given the following:

- The reactor has tripped from 100%.
- Approximately twenty minutes after the trip:
  - Intermediate range channel N35 is reading 1 x $10^{-10}$ amps and steady
  - Intermediate range channel N36 is reading 2 x $10^{-11}$ amps and decreasing slowly
  - Source range channels N31 and N32 are deenergized

N35 is ___(1)___ and the operators must ___(2)___ to energize the Source Range Nuclear Instruments.

A. (1) under-compensated
(2) place both source range "High Flux at Shutdown" switches in the BLOCK position.

B. (1) over-compensated
(2) place both source range "High Flux at Shutdown" switches in the BLOCK position.

C. (1) under-compensated
(2) depress both "P-6 PERMISSIVE DEFEAT 1(2)" pushbuttons.

D. (1) over-compensated
(2) depress both "P-6 PERMISSIVE DEFEAT 1(2)" pushbuttons.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. It is plausible because N35 is under-compensated, but the High Flux at Shutdown switch will not energize the Source Range instruments.

B. Incorrect. It is plausible because the Candidate may not understand under and over compensation and how it relates to IRNI behavior and switch functions.

C. Correct.

D. Incorrect. It is plausible because the P-6 Permissive Defeat 1 and 2 pushbuttons must be depressed, but N35 is under-compensated.

Technical Reference(s): O-2.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 074 EK2.08 | |
| | Importance Rating | 2.5 | |

Knowledge of the interrelations between the following Inadequate Core Cooling: Sensors and detectors

Proposed Question: RO Question # 58

Plant conditions:

- A small break LOCA resulted in a reactor trip and safety injection
- The crew was performing E-1, LOSS OF REACTOR OR SECONDARY COOLANT, when multiple ECCS failures resulted in transition to FR-C.1, RESPONSE TO INADEQUATE CORE COOLING
- The crew is initiating depressurization of SGs to facilitate SI Accumulator injection

Which ONE of the following describes the parameters monitored to determine if the depressurization will be stopped?

A. SG pressure at 160 psig; Thot below 390°F

B. SG pressure at 160 psig; Core Exit Thermocouples below 700°F

C. RCS pressure at 160 psig; Thot below 390°F

D. RCS pressure at 160 psig; Core Exit Thermocouples below 700°F

Proposed Answer: A

Explanation (Optional):

A. Correct. WR Thot is evaluated at less than 390°F so that the depressurization can be terminated until ECCS Accumulators are isolated. SG pressure at 160 psig when depressurization is stopped

B. Incorrect. SG pressure is correct. However, CETs are evaluated throughout FR-C.1 and also as entry conditions to FR-C.1, but not for purposes of RCS depressurization for accumulator injection.

C. Incorrect. Plausible because the pressure value is correct and Thot is also correct, but the pressure is required for SGs, not RCS. The applicant may consider 160 psig in RCS because RCS pressure must be low for accumulator injection.

D. Incorrect. Plausible for same reasons as B and C.

Technical Reference(s): FR-C.1 Step 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RFRC1C, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Meets K/A by requiring identification of the correct detector and the associated value requiring action during a high-level evolution in FR-C.1. While it appears to be a setpoint memorization question, it can be answered by knowing the bases associated with the step.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 061 AK3.02 | |
| | Importance Rating | 3.4 | |

Knowledge of the reasons for the following responses as they apply to the Area Radiation Monitoring (ARM) System Alarms: Guidance contained in alarm response for ARM system

Proposed Question: RO Question # 59

Plant conditions:

- Reactor at 98% power
- NIS Channel N-44 is out of service
- Reactor Engineering is running an incore flux map to determine QPTR
- Annunciator E-24, RMS AREA MONITOR HIGH ACTIVITY, activates
- Monitor R-7, Incore Detection Area, is in alarm
- No other RMS alarm conditions exist

What action, if any, is required to be taken?

A. Notify RP and request a containment air sample.

B. Run an RCS leak rate surveillance to determine if an RCS leak has initiated.

C. None; this is an expected alarm for these conditions.

D. Direct Reactor Engineering to terminate the flux map while the cause is investigated.

Proposed Answer: C

Explanation (Optional):

A. Plausible because this would be a logical action if the flux map was not in progress.

B. Plausible if applicant assumes every containment RMS alarm should be addressed as potential indication of RCS leakage into containment.

C. CORRECT. The sensitivity and setpoint are such that it does alarm when the fission chambers are not in the storage location or reactor.

D. Plausible because RMS alarm response procedures often direct termination of activities

that would interfere with analysis of the situation or may have caused the alarm.

Technical Reference(s): AR-RMS-7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3901C, 4.01 (As available)

Question Source: Bank # C072.0024

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | E02 EA1.3 | |
| | Importance Rating | 3.8 | |

Ability to operate and / or monitor the following as they apply to the (SI Termination) Desired operating results during abnormal and emergency situations.

Proposed Question: RO Question # 60

Plant conditions:

- An RCS leak resulted in a MANUAL SI and CI.
- The crew is now performing ES-1.1, SI TERMINATION, and is restoring letdown.
- The ES-1.1 FOLDOUT requires monitoring subcooling for SI REINITIATION requirements.

Which ONE of the following identifies the required method for monitoring subcooling and the action required if subcooling goes below the minimum value?

A. RCS Subcooling Margin Monitor; Start SI Pumps as necessary

B. RCS Subcooling Margin Monitor; Initiate a MANUAL SI and CI

C. FIG 1.0 - FIGURE MIN SUBCOOLING; Start SI Pumps as necessary

D. FIG 1.0 - FIGURE MIN SUBCOOLING; Initiate a MANUAL SI and CI

Proposed Answer: C

Explanation (Optional):

A. Plausible because the RCS Subcooling Margin Monitor is available for monitoring but there is no alarm associated with the reinitiation setpoint and the FOLDOUT specifically directs the use of FIG 1.0. The second part is correct.

B. Plausible because the RCS Subcooling Margin Monitor is available for monitoring but there is no alarm associated with the reinitiation setpoint and the FOLDOUT specifically directs the use of FIG 1.0.

C. Correct.

D. Plausible because the foldout directs use of Figure 1.0 but SI initiation is not required.

Technical Reference(s): ES-1.1, FOLDOUT Page (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RES11, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Meets K/A by requiring knowledge of indication(s) monitored for subcooling and action required if minimum value is NOT met.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | E13 EA2.2 | |
| | Importance Rating | 3.0 | |

Ability to determine and interpret the following as they apply to the (Steam Generator Overpressure) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question: RO Question # 61

Plant conditions:

- The plant has tripped.
- The crew has entered FR-H.2, Response to Steam Generator Overpressure, based upon a YELLOW condition on the Heat Sink CSF Status Tree.
- SG 'A' pressure is 1150 psig.
- SG 'B' pressure is 1050 psig.
- SG 'A' Narrow Range level is 55%.
- Instrument Air header pressure has been lost.

Which ONE of the following actions will mitigate the steam generator condition in accordance with FR-H.2?

A. Raise SG Blowdown Flow rate.

B. Open MSIV Bypass Valves and/or steam supply to TDAFW pump.

C. Transition to FR-H.3 to reduce pressure by reducing SG level.

D. Place Steam Dump Controller in MANUAL in the STEAM PRESSURE mode, and increase demand.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because raising SG blowdown flow would lower level, but this is not in accordance with FR-H.2

B. Correct. Even without instrument air, the MSIV Bypass Valves can be opened to reduce pressure

C. Incorrect. SG level would have to be >90% to perform this action but it is plausible because it is a step in the procedure.

D. Incorrect. Instrument Air is not available, so steam dumps would also not be available. Plausible because if IA was available, this could possibly be used if MSIS had not occurred.

Technical Reference(s): FR-H.2, Rev 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RFRH2C, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Called new but developed for a Watts Bar exam in 2007 that was not administered

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 069 2.2.39 | |
| | Importance Rating | 3.9 | |

Equipment Control: Knowledge of less than or equal to one hour technical specification action statements for systems.

Proposed Question: RO Question # 62

Plant conditions:

- The unit is returning to service from a refueling outage
- Currently in Mode 4, heating up to Mode 3
- The Shift Manager has just declared the air lock inner door seal inoperable.

Which ONE of the following actions must be performed within one hour?

A. Terminate the heatup and return to Mode 5.

B. Verify the affected air lock outer door is closed.

C. Verify the affected air lock door interlock mechanism is operable.

D. Run the leakage surveillance test on the unaffected air lock to verify operability.

Proposed Answer: B

Explanation (Optional):

A. Plausible because all containment operability requirements are effective in Mode 4. However with the unit already in Mode 4, compliance with the REQUIRED ACTION allows the heatup to continue.

B. CORRECT. ITS 3.6.2, REQUIRED ACTION A.1.

C. Plausible because the interlock affects air lock operability. However, the air lock is already inoperable and the interlock is designed to prevent opening both doors simultaneously; not to compensate for a bad seal.

D. Plausible as similar to the requirements for dual train components (such as EDG). However, the inner and outer doors in each air lock provide their own backup protection,

not the other air lock. In fact, one door could be inoperable in each air lock as long as the REQUIRED ACTION for each one is met.

Technical Reference(s): ITS 3.6.2, REQUIRED ACTION A.1. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 9

55.43

Shielding, isolation, and containment design features, including access limitations.

Comments:

Meets K/A by requiring knowledge of a 1 hour REQUIRED ACTION for a containment integrity issue.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 076 AK3.05 | |
| | Importance Rating | 2.9 | |

Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity : Corrective actions as a result of high fission-product radioactivity level in the RCS

Proposed Question: RO Question # 63

Plant conditions:

- 50% power
- 8 days ago, after a rapid power reduction from 100% power, RCS activity level began rising slowly and has continued to slowly rise
- Radiation Monitor R-9 has gone into high alarm and AR-RM-9, R-9 LETDOWN LINE MONITOR, has directed entry into AP-RCS-3, HIGH REACTOR COOLANT ACTIVITY
- The CRS determines that a plant shutdown and reduction in RCS Tavg is required.

Which ONE of the following describes the reason that a reduction in RCS temperature is required in accordance with AP-RCS.3 and/or Technical Specifications?

A. The RCS must be placed in an operational mode where the TS limit on RCS activity does not apply.

B. RCS temperature must be lowered to comply with accident analysis assumptions for a Main Steam Line Break with TS maximum primary to secondary leak rate.

C. RCS temperature must be below the saturation temperature for the MSSV lift setpoints in case a steam generator tube rupture occurs.

D. Reducing RCS temperature raises the decontamination factor of Letdown System ion exchangers, providing reasonable assurance that 10CFR100 dose limits will not be exceeded in the event of a LOCA.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. The statement by itself is true but is an incorrect answer for the question

proposed.

B. Incorrect. Plausible because it is similar to other TS bases related to SG tube leakage

C. Correct. In the event of a required shutdown, Tavg must be lowered to 500˚F to minimize the chance that an MSSV will open during a SGTR.

D. Incorrect. Plausible because it is similar to discussions about reactivity effects of raising and lowering temperature through an ion exchanger.

Technical Reference(s): TS 3.4.16 Basis (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP17C, 1.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Meets K/A by requiring knowledge of both an operator action in a procedure with only one "hands-on" operation and the basis for a required shutdown and cooldown.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | E10 2.2.37 | |
| | Importance Rating | 3.6 | |

Equipment Control: Ability to determine operability and / or availability of safety related equipment.

Proposed Question: RO Question # 64

Plant conditions:

- A reactor trip occurred with a subsequent loss of off-site power
- Bus 16 tripped on protective relay actuation
- Charging Pump 1A is running
- AOV-427, LETDOWN ISOLATION VALVE, is failed closed
- Excess Letdown is NOT in service
- The crew is performing ES-0.3, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL
- The crew has completed the cooldown to 500°F and are preparing to "Depressurize RCS To 1500 PSIG"

Which ONE of the following describes the component(s) available to depressurize the RCS to 1500 PSIG in accordance with ES-0.3?

A. Auxiliary Spray Valve, AOV-296

B. PRZR PORV-430 ONLY

C. PRZR PORV-431C ONLY

D. PRZR PORV-430 <u>and</u> PRZR PORV-431C

Proposed Answer: B

Explanation (Optional):

A. Plausible because this is the preferred path but is not implemented with Letdown OOS.

B. CORRECT. A procedure NOTE specifies using a PORV with an operable block valve. Power is available to MCC "C" which feeds MOV-516, the block valve for PORV-430.

C. Plausible because the valve numbers and power supplies are somewhat inverted in that MCC "C" feeds MOV-516 and MCC "D" feeds MOV-515.

D. Plausible if applicant does not know that only one PORV is opened if Auxiliary Spray is unavailable.

Technical Reference(s): ES-0.3, NOTE prior to Step 5 and Step 5 P-12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RES03C, 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Meets K/A by having applicant identify the depressurization flowpath that meets the operability requirements of the procedure step in ES-0.3.

Higher order because the applicant must carefully evaluate plant conditions and apply the procedure NOTE to correctly answer the question.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 003 AA1.01 | |
| | Importance Rating | 2.9 | |

Ability to operate and / or monitor the following as they apply to the Dropped Control Rod: Demand position counter and pulse/analog converter

Proposed Question: RO Question # 65

The operating crew is preparing to retrieve a dropped Bank D control rod in accordance with ER-RCC.1, RETRIEVAL OF A DROPPED RCC

Which ONE of the following describes the operation of the associated Group Step Counters and Pulse to Analogue (P/A) converter prior to the rod retrieval?

A. Set only the associated group step counters to ZERO; select the appropriate bank on the P/A converter and reset the digital indication to ZERO.

B. Set only the associated group step counters to ZERO; record the reading on the P/A converter but do NOT change the settings.

C. Set both group step counters in the associated bank to ZERO; select the appropriate bank on the P/A converter and reset the digital indication to ZERO.

D. Set both group step counters in the associated bank to ZERO; record the reading on the P/A converter but do NOT change the settings.

Proposed Answer: A

Explanation (Optional):

A. CORRECT. The P/A Converter feeds only rod control components associated with Control Bank rods and it is manually set to ZERO after the bank position is recorded and before rod withdrawal begins.

B. Plausible because the first part is correct and the setting must be correct. However, it is set to ZERO so that it counts up to the bank position as if the bank were moving.

C. Plausible in that the applicant must understand the function of the P/A converter. Since there is no overlap or insertion alarm associated with the Shutdown Bank rods, the P/A converter is unnecessary. The second part is correct.

D. Plausible in that the applicant must understand the function of the P/A converter. Since there is no overlap or insertion alarm associated with the Shutdown Bank rods, the P/A converter is unnecessary. The second part balances the choices.

Technical Reference(s): ER-RCC.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RER11C, 4.0 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Meets K/A by requiring knowledge of the design application and operation of the P/A Converter during a dropped rod retrieval. The P/A Converter is in the field at GINNA so the control room team is directing someone to set it.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 1 | |
| | K/A # | G1 2.1.5 | |
| | Importance Rating | 2.9 | |

Conduct of Operations: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Proposed Question: RO Question # 66

Plant conditions:

- The date is 6/21/2010.
- The time is 1615.
- You are scheduled to relieve the shift at 1830 this evening.
- You have worked the 1900-0700 shift for the past 5 nights.
- You receive a call from the on-shift CRS to provide emergency relief for the HCO, who was transported to the hospital at 1545 this afternoon.

Which ONE of the following describes (1) the LATEST time that a relief must be on-shift, and (2) whether you may arrive PRIOR to your scheduled shift <u>and</u> COMPLETE your shift WITHOUT violating overtime limitations?

A. (1) 1645
(2) You may relieve the shift early and work your entire shift without violating overtime limitations

B. (1) 1645
(2) You may NOT relieve the shift early and work your entire shift because doing so will violate overtime limitations

C. (1) 1745
(2) You may relieve the shift early and work your entire shift without violating overtime limitations

D. (1) 1745
(2) You may NOT relieve the shift early and work your entire shift because doing so will violate overtime limitations

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because only one overtime limit will be violated, and applicant may not consider it. Additionally, one hour is reasonable to provide shift relief

B. Incorrect. Plausible because the second half is correct and one hour is reasonable to provide shift relief

C. Incorrect. The time is correct for providing relief but the applicant may not relieve the shift and continue through entire shift because they would violate 72 hours in 7 day requirements

D. Correct. Would violate the rule for 72 hours in 7 days

Technical Reference(s): TS 5.2.2
CNG-SE-1.01-1002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 1 | |
| | K/A # | G1 2.1.8 | |
| | Importance Rating | 3.4 | |

Conduct of Operations: Ability to coordinate personnel activities outside the control room.

Proposed Question: RO Question # 67

Plant conditions:

- A station blackout has occurred.
- The crew is performing actions of ECA-0.0, Loss of All AC Power.
- Off-Site power has NOT been restored.
- BOTH EDGs are tripped and have NOT been restarted.
- SGs depressurization is in progress.

Which ONE of the following actions may be taken to provide power to a safeguards bus, and the procedure used following the power restoration?

Direct an AO to align the TSC Diesel to...

A. Bus 14; the bus is considered restored when power is aligned and the crew may transition to the appropriate recovery procedure when directed.

B. Bus 14; the bus is NOT considered restored when power is aligned and the crew must remain in ECA-0.0 until another power source is aligned to a safeguards bus.

C. Bus 16; the bus is considered restored when power is aligned and the crew may transition to the appropriate recovery procedure when directed.

D. Bus 16; the bus is NOT considered restored when power is aligned and the crew must remain in ECA-0.0 until another power source is aligned to a safeguards bus.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. TSC diesel is aligned to opposite safeguards bus. Although the bus would be energized, it is NOT considered restored IAW ECA-0.0

B. Incorrect. second part is correct but bus 16 is restored, not bus 14

C. Incorrect. Correct bus identified, but bus is not considered restored, and crew must remain in ECA-0.0 until the bus is restored for an emergency DG

D. Correct. Note prior to ECA-0.0, Step 27

Technical Reference(s): ECA-0.0, note prior to step 27 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R0801C, Obj 1.10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 2 | |
| | K/A # | G2 2.2.37 | |
| | Importance Rating | 3.6 | |

Equipment Control: Ability to determine operability and / or availability of safety related equipment.

Proposed Question: RO Question # 68

Plant conditions:

- The plant is operating at 100% power.
- A loss of DC Bus 'A' occurs.

Which ONE of the following describes equipment that is made INOPERABLE by this failure?

A. All Steam Dump Valves

B. Pressurizer PORV 431C

C. Train 'A' SI Actuation Circuitry

D. Turbine Driven AFW Pump

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Losing DC Bus 'B' would make this true. Losing DC Bus 'A' will fail several valves, and the remainder will only go full open or closed

B. Incorrect. PORV 430 is inoperable, but PORV 431C is powered by Train B

C. Correct. DC Power required for SI actuation circuitry

D. Incorrect. If DC Bus B was lost, the TDAFW Discharge MOV would fail as is, but Train A loss does not make TDAFW inoperable

Technical Reference(s): ER-ELEC.2, Attachment 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R0901C, Obj 1.06 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Comprehension because it requires understanding not only of power supplies, but the effect it has on the components that receive power

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 2 | |
| | K/A # | G2 2.2.35 | |
| | Importance Rating | 3.6 | |

Equipment Control: Ability to determine Technical Specification Mode of Operation.

Proposed Question: RO Question # 69

Plant conditions:

- The plant is shutdown, preparing for refueling.
- Maintenance is preparing to de-tension reactor vessel head closure bolts.
- All systems are normally aligned for the current plant conditions.
- At 0600, the following conditions are noted:
  - Cold Leg WR Temperature indicators are 175°F.
  - WR RCS Pressure indicators are 100 psig.

Subsequently, RHR is lost and ALL Cold Leg WR Temperature indicators begin rising at 2°F/minute.

Based on the above indications and assuming the above conditions and trends continue, which ONE of the following correctly identifies the plant MODE at 0600 and MODE at 0700?

| | MODE at 0600 | MODE at 0700 |
|---|---|---|
| A. | MODE 5 | MODE 3 |
| B. | MODE 5 | MODE 4 |
| C. | MODE 6 | MODE 4 |
| D. | MODE 6 | MODE 5 |

Proposed Answer: B

Explanation (Optional):

A. Incorrect. At 0600, plant is initially in Mode 5 (RCS temp < or equal to 200°F, Keff < 0.99). At 0700, RCS temperature will be 295°F, which is Mode 4 (RCS temp < 350°F. Plausible because the applicant may either calculate incorrectly or confuse modes

B. Correct. Plant is in Mode 5 initially. Plant is in Mode 4 starting at approximately 0613

C. Incorrect. Plant is in Mode 5 initially. Plant would be in Mode 6 initially if one or more reactor vessel head bolts was less than fully tensioned and will enter Mode 4 at 0613 if current trends continue.

D. Incorrect. Plant is in Mode 5 initially. Plant would be in Mode 6 initially if one or more reactor vessel head bolts was less than fully tensioned.

Technical Reference(s): Technical Specification table 1.1-1, Definitions (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

Question Source: Bank #

Modified Bank # WTSI 65657 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Question meets KA. Question requires examinee ability to determine Mode of Operation based on plant indications and procedures.

Modified from Braidwood 2007 NRC exam. Modified heatup rate from 3 to 2 to change correct answer

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 3 | |
| | K/A # | G3 2.3.14 | |
| | Importance Rating | 3.4 | |

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: RO Question # 70

Plant conditions:

- A Steam Generator Tube Rupture has occurred on SG "B".
- The crew has performed all actions of E-3, Steam Generator Tube Rupture, up to the step to commence depressurization of the RCS.
- All equipment is functioning as designed.

Which ONE of the following describes the status of "B" SG Atmospheric Relief Valve, and the reason for the status?

A. CLOSED with controller in Manual; prevent radioactive release to atmosphere.

B. CLOSED with controller in Manual; ensures minimum RCS subcooling will be maintained when RCS depressurization is initiated.

C. Set at 1050 psig with controller in AUTO; prevent uncontrolled radioactive release due to SG safety valve lifting.

D. Set at 1050 psig with controller in AUTO; ensures minimum RCS subcooling will be maintained when RCS depressurization is initiated.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because it is logical to maintain valve closed but controller will not be in manual. Reason is correct

B. Incorrect. Same reason as option A, and additionally, reason is plausible because if the ARV stuck open on a ruptured SG, the depressurization would also cause depressurization of the RCS. This would result in loss of RCS subcooling.

C. Correct. Controller is in AUTO which allows valve to open as required if pressure rises. This prevents safety valves from lifting and potentially becoming stuck open, causing radioactive release.

D. Incorrect. Correct for status of valve, but reason is incorrect. Plausible because valve would be placed in manual and closed if it stuck open below 1050 psig, but this is not the reason that the valve is placed in AUTO. The remainder of the steps for SG isolation are correct for this reason.

Technical Reference(s): E-3, Rev 46, step 3
Westinghouse Setpoint Document (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: REP03C, Obj 1.03, 2.01 (As available)

Question Source: Bank # WTSI 64973
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Wolf Creek 2009

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 3 | |
| | K/A # | G3 2.3.13 | |
| | Importance Rating | 3.4 | |

Radiation Control: 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.

Proposed Question: RO Question # 71

Plant conditions:

- The unit is at 90% power.
- All systems are in their normal alignment.
- Auxiliary Building (AB) Ventilation IMS Switch is in " FILTER IN".
- "C" Gas Decay Tank (GDT) is being released.

1. What is the effect if R-13, Auxiliary Building Particulate Monitor, or R-14, Auxiliary Building Gas Monitor, goes into high alarm?

and

2. What are the operator actions for that plant response?

A.
1. If R-13, goes into high alarm, "C" Gas Decay Tank release is automatically secured.
2. Verify the GDT release AOV-RCV-014 to the plant vent closes.
   Ensure 1F AB Exhaust fan is no longer running.
   Ensure 1A, 1B and 1C Intermediate Building Exhaust fans are no longer running

B.
1. If R-13, goes into high alarm, "C" Gas Decay Tank release is automatically secured.
2. Verify the GDT release AOV-RCV-014 to the plant vent closes.
   Ensure 1A, 1B, 1C and 1F AB Exhaust fans are no longer running.
   Ensure 1A, 1B and 1C Intermediate Building Exhaust fans are no longer running.

C.
1. If R-14, goes into high alarm, "C" Gas Decay Tank release is automatically secured.
2. Verify the GDT release AOV-RCV-014 to the plant vent closes.
   Ensure 1F AB Exhaust fan is no longer running.
   Ensure 1A, 1B and 1C Intermediate Building Exhaust fans are no longer running.

D.
1. If R-14, goes into high alarm, "C" Gas Decay Tank release is automatically secured.
2. Verify the GDT release AOV-RCV-014 to the plant vent closes.
   Ensure 1A, 1B, 1C and 1F AB Exhaust fans are no longer running.
   Ensure 1A, 1B and 1C Intermediate Building Exhaust fans are no longer running.

Proposed Answer: C

Explanation (Optional):

A. R-13 will cause this realignment of the AB exhaust system but it will not stop the Gas Decay Tank release so RCV-14 (the GDT release AOV to the plant vent) will not close

B. R-13 will cause a realignment of the AB exhaust system but it will not stop the Gas Decay Tank release so RCV-14 (the GDT release AOV to the plant vent) will not close. This is correct if the Auxiliary Building (AB) filters are "OUT", but in the stem the Auxiliary Building (AB) filters are "IN", so the additional fans are not affected

C. With a high alarm on R-14 and the AB filters are "IN" the automatic actions that should occur are: RCV-14 (the GDT release AOV to the plant vent) closes. 1F AB Exhaust fan receives a trip signal. 1A, 1B and 1C Intermediate Building fans receive trip signals

D. With a high alarm on R-14, RCV-14 (the GDT release AOV to the plant vent) closes. This is correct if the Auxiliary Building (AB) filters are "OUT", but in the stem the Auxiliary Building (AB) filters are "IN", so the additional fans are not affected

Technical Reference(s): LP R3901C, Radiation Monitoring System pg. 18
LP R3801C, Waste Disposal System pg. 25 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3901C Obj 1.03 (As available)

Question Source: Bank # WTSI 65980
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2008 Ginna

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 4 | |
| | K/A # | G4 2.4.30 | |
| | Importance Rating | 2.7 | |

Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Proposed Question: RO Question # 72

Plant conditions:

- A reactor trip occurred from 100% power.
- The crew is performing ES-0.1, Reactor Trip Response.
- Due to equipment failures following the trip, the Shift Manager declares an Unusual Event .
- There is NO radiological concern or release in progress or anticipated.

Which ONE of the following describes whether or not the local and state authorities must be notified, and the reason why?

State and Local authorities...

A. must be notified because an unplanned reactor trip has occurred.

B. must be notified because the event resulted in declaration of an Emergency Classification.

C. are NOT required to be notified because there is no radioactive release in progress or anticipated.

D. are NOT required to be notified because local and state notifications are only required at the ALERT level or above.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the NRC would be notified within 4 or 8 hours depending on the event and level of equipment malfunction, if there was no classification

B. Correct. State and Local notifications are required within 15 minutes for emergency

classifications

C. Incorrect. Plausible because with no release in progress, the applicant may believe that since evacuation or shelter are not required, that local authorities do not require notification.

D. Incorrect. Plausible because the entire ERO is not required to be manned unless the classification is alert or above, but state and local must always be notified

Technical Reference(s): EPIP 1-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 4 | |
| | K/A # | G4 2.4.39 | |
| | Importance Rating | 3.9 | |

Emergency Procedures / Plan: Knowledge of the RO's responsibilities in emergency plan implementation.

Proposed Question: RO Question # 73

Plant conditions:

- The plant is in a General Emergency.
- The Ginna Station Nuclear Emergency Response Plan is being implemented.

From the list below, which one of the following correctly identifies the responsibilities of the HCO/CO during the General Emergency in accordance with EPIP 5-7, Emergency Organization?

1. Report all unusual observations or communications to the Shift Manager.
2. Sound the alarm and make announcements as necessary.
3. Peer check the Shift Manager in the diagnosis of unusual event conditions and above.
4. Check that the Control Room ventilation system is in re-circulation mode.
5. Assist as directed and inform Shift Manager of all Control Room changes.
6. Ensure search and rescue is initiated, if necessary, per EPIP 1-8.

A. 1, 3, 5

B. 2, 4, 6

C. 1, 3, 6

D. 2, 4, 5

Proposed Answer: D

Explanation (Optional):

A. Incorrect. # 1 is the responsibility of the C/R Communicator. # 3 is the responsibility of the STA per EPIP 5-7, Emergency Organization

B. Incorrect. # 6 is the responsibility of the Emergency Coordinator per EPIP 5-7, Emergency Organization

C. Incorrect. # 1 is the responsibility of the C/R Communicator. # 3 is the responsibility of the STA per EPIP 5-7, Emergency Organization.. # 6 is the responsibility of the Emergency Coordinator.

D. Correct Per EPIP 5-7, Emergency Organization

Technical Reference(s): EPIP 5-7, Emergency Organization (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: N/A (As available)

Question Source: Bank # WTSI 65965

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Ginna 2008

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 2 | |
| | K/A # | G2 2.2.13 | |
| | Importance Rating | 4.1 | |

Equipment Control: Knowledge of tagging and clearance procedures.

Proposed Question: RO Question # 74

Which ONE of the following jobs would **REQUIRE** an Isolated Work Area (a TAGOUT), due to its meeting the definition of a "Hazardous Energy" in accordance with CNG-OP-1.01-1007, Clearance and Safety Tagging? (Assume **NO** instrument calibrations will be performed.)

Work on a(n) ...

A. AC circuit where the maximum voltage is 45 volts AC.

B. DC circuit where the maximum voltage is 40 volts DC.

C. Hydraulic system which has a maximum pressure of 100 psig and a maximum temperature of 75°F.

D. Hydraulic system which has a maximum pressure of 40 psig and a maximum temperature of 110°F.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Per procedure a hold is required for > 50 volts (AC OR DC).

B. Incorrect. Per procedure a hold is required for > 50 volts (AC OR DC).

C. Correct. Per procedure a hold is required for > 50 psig hydraulic pressure and/or > 120°F temperature

D. Incorrect. Per procedure a hold is required for > 50 psig hydraulic pressure and/or > 120°F temperature

Technical Reference(s): CNG-OP-1.01-1007 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: N/A (As available)

Question Source: Bank #

Modified Bank # WTSI 65954 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

I didn't have this procedure, just changed the question based upon the information in the existing question

Modified from a 2008 Ginna NRC exam question

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 1 | |
| | K/A # | G1 2.1.3 | |
| | Importance Rating | 3.7 | |

Conduct of Operations: Knowledge of shift or short-term relief turnover practices.

Proposed Question: RO Question # 75

Given the following:

- The time and date is 0630, June 25, 2010.
- You are the on-coming HCO performing shift turnover.
- The last shift you worked was June 21, 2010 from 0600 to 1800.

In accordance with OPS-SHIFT-TURNOVER, prior to assuming the shift, you must review shift logs (official records) back to AT LEAST 0630 on...

A. June 24

B. June 23

C. June 22

D. June 21

Proposed Answer: ~~B~~ D correct, per revised key. J. Fish

Explanation (Optional):

A. Incorrect. June 24 would represent 24 hours, and the MINIMUM review is 48 hours or back to the last time you were on shift, whichever is less. Plausible because it is reasonable that 24 hours could be the standard

B. Correct. 2 days is 48 hours, which is less time than the last time you were on shift

C. Incorrect. Time represents 72 hours, which is plausible because it falls directly between the 2 times identified in the procedure

D. Incorrect. Plausible because it is the last time you were on shift, which is one of the 2 times identified in the procedure.

Technical Reference(s): CNG-OP-1.01-1002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 056 | AA2.23 |
| | Importance Rating | | 3.9 |

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Turbine trip-reactor button and indicator

Proposed Question: SRO Question # 76

Plant conditions:

- The plant is at 100% power.
- A reactor trip occurred
- Subsequent to the trip, a loss of off-site power occurred.
- Turbine stop valve position cannot be verified.
- The CO manually depresses the Turbine Trip pushbutton.
- He reports that he still cannot determine whether the turbine has tripped.

Which ONE of the following describes the action that will be required next, and if conditions require the plant to be placed in Mode 5, which procedure will be used to perform RCS cooldown to RHR entry conditions?

A. Shut MSIVs; ES-0.2, Natural Circulation Cooldown.

B. Shut MSIVs; O-2.2, Plant Shutdown From Hot Shutdown to Cold Conditions.

C. Manually run back the turbine or trip EHC Pumps; ES-0.2, Natural Circulation Cooldown.

D. Manually run back the turbine or trip EHC Pumps; O-2.2, Plant Shutdown From Hot Shutdown to Cold Conditions.

Proposed Answer: A

Explanation (Optional):

A. Correct. The crew will go from E-0 to ES-0.1, then transition to ES-0.2 until RHR entry conditions. O-2.2 can be used once Mode 5 is reached

B. Incorrect. Closing MSIVs is correct but O-2.2 will not be used until after exit from ES-0.2

Incorrect. Plausible because these methods will also result in a turbine shutdown.

C. Correct procedure

D. Incorrect. Plausible same reason as option C but wrong procedure in this option

Technical Reference(s): E-0, step 2 RNO
E-0, step 20 RNO (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: REP00, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 055 | EA2.04 |
| | Importance Rating | | 4.1 |

Ability to determine or interpret the following as they apply to a Station Blackout: Instruments and controls operable with only dc battery power available

Proposed Question: SRO Question # 77

Plant conditions:

- The plant was at 100% power.
- A loss of all off site power occurred approximately 20 minutes ago.
- Both "A" and "B" Diesel Generators failed to start and cannot be started.

Which ONE of the following Control Room controls or indications will remain usable to control the initial response and describe the reporting requirements to offsite authorities.?

A. Microprocessor Rod Position Indication (MRPI)
First report to off-site authorities is required within 15 minutes of initial emergency classification

B. Steam Generator ARV Controllers
First report to off-site authorities is required within 15 minutes of initial emergency classification

C. Microprocessor Rod Position Indication (MRPI)
First report to off-site authorities is required within 60 minutes of initial emergency classification

D. Steam Generator ARV Controllers
First report to off-site authorities is required within 60 minutes of initial emergency classification

Proposed Answer: B

Explanation (Optional):

A. Incorrect - MCC-1K supplies power to MRPI and will be de-energized, and state and local authorities must be notified within 15 minutes

B. Correct - ARVs are available for use as stated in ECA-0.0, and 15 minutes is the required first call

C. Incorrect - MCC-1K supplies power to MRPI and will be de-energized, and 60 minutes is plausible because the NRC is required to be notified in that time

D. Incorrect. ARVs are available, and also plausible on time for same reason as option C

Technical Reference(s): ECA-0.0
EPIP 1-0
EPIP 1-5
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R3101C, Obj 1.02
REC00C, Obj 2.01
(As available)

Question Source: Bank # WTSI 62905
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Callaway 2009

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

10 CFR Part 55 Content: 55.41
55.43 1
Conditions and limitations in the facility license

Comments:
changed distracter options but stem essentially left intact.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 026 | AA2.06 |
| | Importance Rating | | 3.1 |

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The length of time after the loss of CCW flow to a component before that component may be damaged

Proposed Question: SRO Question # 78

Initial conditions:

- The plant is operating at 100% power.
- Component Cooling Water temperatures have been rising, and the CRS is addressing TS 3.7.7 for operability of the Component Cooling Water System.
- To comply with the action statement of TS 3.7.7, the CRS determines that a plant shutdown to Mode 3 is required.

Current conditions:

- A loss of Component Cooling Water has occurred and the crew is performing actions of AP-CCW.2, Loss of CCW During Power Operation.
- The CCW System has been without flow for 2 minutes.
- RCP motor bearing temperatures are 192°F and rising.

Which ONE of the following describes the action required, and assuming operability cannot be restored to CCW, the effect on compliance with TS 3.7.7?

A. Trip the reactor, trip the RCPs, and perform E-0, Reactor Trip or Safety Injection; All requirements of TS 3.7.7 are met when the plant is in Mode 3.

B. Remain in AP-CCW.2 and continue attempts to restore CCW; All requirements of TS 3.7.7 are met when the plant is in Mode 3.

C. Trip the reactor, trip the RCPs, and perform E-0, Reactor Trip or Safety Injection; All requirements of TS 3.7.7 will NOT be met until the plant is in Mode 4.

D. Remain in AP-CCW.2 and continue attempts to restore CCW; All requirements of TS 3.7.7 will NOT be met until the plant is in Mode 4.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. TS 3.7.7 requires the unit to be in Mode 4 within 12 hours if operability cannot be restored.

B. Incorrect. Plausible because RCP temperatures have not reached the trip setpoint yet. Action is correct for operability

C. Correct. TS 3.7.7 requires continuing to Mode 4. RCPs have been without CCW for over 2 minutes

D. Incorrect. Plausible same as Option B

Technical Reference(s): AP-CCW.2
TS 3.7.7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP02C, Obj 2.01 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 062 | G2.4.4 |
| | Importance Rating | | 4.7 |

Emergency Procedures / Plan: Ability to recognize abnormal indication for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: SRO Question # 79

Plant conditions:

- The plant is operating at 20% power, with startup to 100% power in progress
- Main feedwater pump B is running
- The main generator has just been synchronized to the grid and turbine load is being held at 50 MWe to meet turbine minimum load hold time requirements
- Offsite power is in a 50/50 Normal configuration
- Busses 11A & 11B have not yet been transferred to Aux Unit Transformer 11
- Subsequently, Transformer 7T experiences an internal differential fault and trips (lockout relays actuated)

Based upon the conditions indicated, determine which of the following describes (1) the plant response, and (2) the crew actions required for this set of conditions?

| | **Plant Response** | **Crew Actions** |
|---|---|---|
| A. | • Bus 12B is deenergized<br>• Buss 11A & 11B remain energized<br>• D/G B starts and reenergizes Busses 16 & 17 | • Enter AP-ELEC.1, Loss of 12A and/or 12B Busses<br>• Manually trip the reactor and enter E-0, Reactor Trip or Safety Injection |
| B. | • Bus 12B is deenergized<br>• Buss 11A & 11B remain energized<br>• D/G B starts and reenergizes Busses 16 & 17 | • Verify automatic reactor trip and enter E-0, Reactor Trip or Safety Injection<br>• Restore power to bus 12B using ER-ELEC.1, Restoration of Off-Site Power, when directed by E-0. |
| C. | • Bus 12A is deenergized<br>• Buss 11A is deenergized, 11B remains energized<br>• D/G A starts and reenergizes Busses 14 & 18 | • Enter AP-ELEC.1, Loss of 12A and/or 12B Busses<br>• Manually trip the reactor and enter E-0, Reactor Trip or Safety Injection |

D.
- Bus 12A is deenergized
- Buss 11A is deenergized, 11B remains energized
- D/G A starts and reenergizes Busses 14 & 18

- Verify automatic reactor trip and enter E-0, Reactor Trip or Safety Injection
- Restore power to Bus 12A using ER-ELEC.1, Restoration of Off-Site Power, when directed by E-0.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Part (2) is correct, but the examinee must recognize that since 11A and 11B have not been aligned to the Bus 12A, Bus 11A will not remain energized.

B. Incorrect. The examinee must recognize that since 11A and 11B have not been aligned to the Bus 12A, Bus 11A will not remain energized. Also, with power <P-8 setpoint, auto reactor trip will not occur on the loss of a single RCP.

C. Correct. The given conditions indicate a fault on the 7T transformer, which, in the 50/50 Normal alignment, deenergizes Bus 12A. Since Busses 11A & 11B have not yet been aligned to the Bus 12A, Bus 11A is deenergized. Loss of Bus 11A will result in the loss of Busses 14 & 18, an automatic start of D/G A, and the loss of A RCP. Since reactor power is below the P-8 single loop low flow trip setpoint of 25%, the reactor will not receive an auto reactor trip signal. Entry to AP-ELEC.1 is appropriate, and step 3 will address RCP status and require a Manual reactor trip and transition to E-0.

D. Incorrect. Although part (1) is correct, the examinee must recognize that the given initial conditions are below the automatic reactor trip setpoint for single loop operation of 25%.

Technical Reference(s): AP-SW.1, Step 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP19C, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Meets KA because the headers are split locally

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | E12 2.2.22 | |
| | Importance Rating | | 4.7 |

Equipment Control: Knowledge of limiting conditions for operations and safety limits.

Proposed Question: SRO Question # 80

The crew is performing actions of ECA-2.1, Uncontrolled Depressurization of Both Steam Generators.

Which ONE of the following describes the Technical Specification implications of the event?

A. RCS cooldown rates above 100°F per hour may potentially result in brittle fracture of the reactor vessel.

B. Pressurizer cooldown rates above 100°F per hour may potentially result in pressurized thermal shock to the pressurizer.

C. Loss of SG inventory may ultimately result in the RCS Pressure Safety Limit being exceeded.

D. Loss of SG inventory may ultimately result in the Reactor Core Safety Limit being exceeded.

Proposed Answer: A

Explanation (Optional):

A. Correct. Concern about non-ductile failure (brittle fracture) of reactor vessel is basis for maximum cooldown rates.

B. Incorrect. PTS requires neutron embrittlement and pressurizer does not receive embrittlement. Plausible because PTS is a concern related to cooldown rates, and the pressurizer does have administrative restrictions on cooldown rates, just not as limiting as the RCS.

C. Incorrect. Plausible because ECA-2.1 is associated with loss of SG inventory because there is an uncontrolled loss of SG mass. However, ECA-2.1 accounts for this by directing action for AFW. Also, if SG inventory was lost. the RCS pressure safety limit could be challenged due to lack of heat removal.

D. Incorrect. Plausible because the applicant must know the factors affecting the RCS Core Safety Limit, and RCS temperature is a factor. A loss of SG inventory will result in a rise in RCS temperature. However, this safety limit is associated with plant operation in Mode 1.

Technical Reference(s): TS 3.4.3
PTLR (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 025 2.4.3 | |
| | Importance Rating | | 3.9 |

Emergency Procedures / Plan: Ability to identify post-accident instrumentation.

Proposed Question: SRO Question # 81

Plant conditions:

- The plant is in Mode 5.
- RCS temperature is 185°F.
- A loss of RHR has occurred due to trip of BOTH RHR Pumps.
- Pressurizer Level is 14% and lowering at approximately 1% per minute.
- RCS temperature is rising at approximately 2°F per minute.
- The Shift Manager is evaluating EPIP-1.0 to determine the Emergency Classification.

(1) Which ONE of the following instruments will be used to determine the appropriate emergency classification for this event?
(2) What are the EAL consequences, if any, if these conditions persist?

(Reference Provided)

A. (1) Core Exit Thermocouples.
(2) Event currently meets the criteria for EAL classification.

B. (1) Core Exit Thermocouples.
(2) Event does NOT currently require classification but will meet the criteria for EAL classification in less than 10 minutes.

C. (1) Pressurizer Level.
(2) Event currently meets the criteria for EAL classification.

D. (1) Pressurizer Level.
(2) Event does NOT currently require classification but event will meet the criteria for EAL classification in less than 10 minutes.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Classification will be required when RCS temperature reaches 200. CETs are post-accident instrumentation

B. Correct.

C. Incorrect. PRZR level helps diagnose an event, but RVLIS is used for classification on RCS inventory

D. Incorrect. PRZR level helps diagnose an event, but RVLIS is used for classification on RCS inventory

Technical Reference(s): EPIP-1.0, EAL 7.2.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | 036 | AA2.03 |
| | Importance Rating | | 4.2 |

Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Magnitude of potential radioactive release

Proposed Question: SRO Question # 82

In accordance with Technical Specifications, which ONE of the following describes plant parameters with LCOs that are designed to limit iodine release and off-site dose following a postulated Fuel Handling Accident?

A. Refueling cavity level ONLY

B. RCS boron concentration ONLY

C. Both RCS temperature and Refueling Cavity level

D. Both RCS temperature and RCS boron concentration

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. Plausible because neutrons are absorbed by boron, but boron concentration concern is inadvertent criticality, not rad release from fuel handling accident

C. Incorrect. Plausible because there is a temperature limitation in Mode 6

D. Incorrect. Same as C

Technical Reference(s): TS 3.9.6 basis (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 7

Fuel handling facilities and procedures.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | 033 AA2.12 | |
| | Importance Rating | | 3.1 |

Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Maximum allowable channel disagreement

Proposed Question: SRO Question # 83

Plant conditions:

- Plant startup in progress.
- N31 and N32 are indicating $2X10^4$ CPS
- N35 indicates $1.5X10^{-10}$ amps
- N36 indicates $7.0X10^{-9}$ amps
- N41, 42, 43, 44 indicate 0%

Which ONE of the following describes the operability of the NIs, and the actions, if any, that are necessary?

A. Source Range and Intermediate Range NIs are operable, since Power Range indicates <8%. NO Technical Specification action is required. A CR must be initiated for the deviation between Intermediate Range Instruments.

B. Intermediate Range NI deviation is outside of tolerance but will be deenergized when Power Range NIs indicate >8%. NO Technical Specification action is required, but a CR must be initiated.

C. N35 is reading too low for the current Source Range level. Within 2 hours, reactor power must be raised OR lowered to a value where Intermediate Range instruments are not required.

D. N36 is reading too high for the current Source Range level. Within 2 hours, reactor power must be raised OR lowered to a value where Intermediate Range instruments are not required.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because a CR will be written, but IR is required below P-10.

B. Incorrect. Plausible because SR de-energizes >P-6, but IR does not. Operability requirements change >10% power.

C. Incorrect. For the current SR indication, IR should be significantly less than N36.

D. NI-4 is approximately 1 decade high for the current source range level. Maximum disagreement for IR detectors is 5% of scale

Technical Reference(s): TS-3.3.1
R3301C, T43C-006A (Attach if not previously provided)
O-6.13

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 58467
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | E03 | 2.4.30 |
| | Importance Rating | | 4.1 |

Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Proposed Question: SRO Question # 84

Plant conditions:

- A LOCA is in progress.
- The crew is initiating an RCS cooldown in accordance with ES-1.2, Post LOCA Cooldown and Depressurization.
- The Shift Manager has completed all required local, state, and NRC notifications for an initial Emergency Classification.

Which ONE of the following describes the additional reporting required to the NRC, if any, beyond the reports already completed?

(Reference Provided)

A. An additional ONE hour report is required ONLY.

B. An additional FOUR hour report is required ONLY.

C. Additional ONE hour AND FOUR hour reports are required.

D. NO additional NRC reports are required because an emergency classification is in effect

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Reportability procedure requires one hour for classification, but not for ECCS discharge or reactor trip

B. Correct.

C. Incorrect. Plausible because a four hour report is required, but the one hour report has been completed for EP

D. Incorrect. Plausible because a report has been made, but there are 4 hour reports that also must be made

Technical Reference(s): CNG-NL-1.01-1004 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: CNG-NL-1.01-1004

Learning Objective: N/A (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1

Conditions and limitations in the facility license

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | E14 2.4.6 | |
| | Importance Rating | | 4.7 |

Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.

Proposed Question: SRO Question # 85

Plant conditions:

- A DBA LOCA has occurred.
- Component Cooling Water has been lost.
- The crew was required to perform ECA-1.1, Loss of Emergency Coolant Recirculation.
- The Crew is now entering FR-Z.1, Response to High Containment Pressure.
- Containment pressure is 31 psig and STABLE.
- BOTH Containment Spray Pumps are OFF.

Which ONE of the following describes the strategy for reducing Containment Pressure?

A. OPERATE Containment Spray Pumps in accordance with the guidance in ECA- 1.1, as directed by FR-Z.1. Continue in FR-Z.1 until exit criteria is met.

B. START both Containment Spray Pumps in accordance with FR-Z.1. RED CSF conditions take precedence over ECA actions.

C. Perform ONLY the FR-Z.1 actions that do NOT conflict with or undo the action taken in ECA-1.1. Two Containment Recirc Coolers will provide adequate depressurization to meet the Containment Safety Function requirements.

D. Do NOT perform actions of FR-Z.1 until the RWST LO-LO level alarm is clear and Containment Spray Pumps may be restarted. Ensure all other automatic actions related to containment isolation have occurred as required.

Proposed Answer: A

Explanation (Optional):

A. Correct. If Containment Spray Pumps have been shut down in ECA-1.1, the action in FR-Z.1 is subordinate to ECA-1.1 actions due to limited inventory for spray and ECCS

B. Incorrect. Plausible because red conditions typically do take precedence over ECA

actions. This is an exception

C. Incorrect. Plausible because Containment Cooling is adequate with 2 coolers, but not containment depressurization under emergency conditions.

D. Incorrect. Plausible because with RWST LO-LO level alarm, there may not be enough NPSH for spray pumps to take a suction on the tank.

Technical Reference(s): FR-Z.1, note prior to step 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RFRZ01C, Obj 2.01 (As available)

Question Source: Bank # WTSI 55223

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Callaway 2005 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 026 | A2.04 |
| | Importance Rating | | 4.2 |

Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of spray pump

Proposed Question: SRO Question # 86

Plant conditions:

- Reactor was manually tripped during a loss of Service Water.
- Service Water CANNOT be recovered at this time.
- The crew transitioned to ES-0.1, Reactor Trip Response
- Subsequently, a LOCA occurred.
- After transition back to E-0, Reactor Trip or Safety Injection, Containment Spray actuates.
- 'A' Containment Spray Pump control switch has a TRIP indication.
- Train 'A' Containment Spray flow is 0 gpm.
- Containment pressure indicates 29 psig and slowly rising.

Which ONE of the following describes the action that will be required?

A. Enter FR-Z.1, Response to High Containment Pressure, due to a RED Path.

B. Enter FR-Z.1, Response to High Containment Pressure, due to an ORANGE Path.

C. Remain in E-0, Reactor Trip or Safety Injection, and use the Continuous Action step to attempt to start 'A' Containment Spray Pump.

D. Continue in E-0 and operate Containment Spray in accordance with E-1, Reactor or Secondary Coolant when that procedure is in effect.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Procedure is correct but Containment is in an ORANGE, not RED, condition

B. Correct. Have already left E-0 to go to ES-0.1, so CSF Status Tree Monitoring is in effect.

C. Incorrect. The step to initiate spray is a CA step, but because CSF STs are in effect, the crew will go to FR-Z.1

D. Incorrect. If already in E-1, the crew would follow that guidance as directed in the note prior to step 1 of FR-Z.1

Technical Reference(s): A-503.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RFRZ1C, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 059 | A2.04 |
| | Importance Rating | | 3.4 |

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: Feeding a dry S/G

Proposed Question: SRO Question # 87

Plant conditions:

- A loss of Feedwater has occurred.
- The crew is performing actions of FR-H.1, Response to Loss of Secondary Heat Sink.
- SG Wide Range levels are 45 inches and lowering.
- Bleed and Feed has been initiated, but only one train of SI is operating.
- Core Exit thermocouples are approximately 590°F and slowly lowering.
- The crew has been able to restore one Main Feedwater Pump.
- Attachment 22, Attachment Restoring Feed Flow, is being performed.

Which ONE of the following describes how feedwater will be initiated in accordance with Attachment 22, and which procedure transition will be made when heat sink is restored and Bleed and Feed is terminated?

A. Initiate flow at less than or equal to 100 gpm; Transition to ES-1.1, SI Termination.

B. Initiate flow at less than or equal to 100 gpm; Transition to E-0, Reactor Trip or Safety Injection.

C. Initiate flow at maximum rate: Transition to ES-1.1, SI Termination.

D. Initiate flow at maximum rate: Transition to E-1, Loss of Reactor or Secondary Coolant.

Proposed Answer: A

Explanation (Optional):

A. Correct. With no further complications, the crew will ensure PORVs are closed and SI is shut down, then go to ES-1.1 at end of FR-H.1

B. Incorrect. Flow is right but procedure is wrong. Will go to E-1 if can't isolate any bleed

path. E-0 would be correct if directed to transition to procedure and step in effect, and original transition was directly from E-0

C. Incorrect. Maximum flow would be initiated to only ONE SG if RCS temperatures were rising. Correct procedure

D. Incorrect. Maximum flow would be initiated to only ONE SG if RCS temperatures were rising.

Technical Reference(s): ATT-22.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RFRH1C, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Part of this was drawn from bank questions, but the transition is added. parameters and conditions new and specific to facility

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 010 2.2.38 | |
| | Importance Rating | | 4.5 |

Equipment Control: Knowledge of conditions and limitations in the facility license.

Proposed Question: SRO Question # 88

Plant conditions:

- The plant is at 100% power.
- Engineering reported that BOTH Pressurizer Safety Valves were set incorrectly during the last Refueling outage.
- Each valve is set to Relieve at 2572 psig.

Which ONE of the following describes the MAXIMUM amount of time allowed to place the Unit in Mode 3 in accordance with Technical Specifications, and the basis for the TS LCO?

A. Six (6) hours; operability of the PRZR Safety valves ensures that RCS pressure is limited to less than 120% of the Safety Limit for RCS pressure for ALL anticipated transients with the EXCEPTION of an RCP locked rotor.

B. Six (6) hours; operability of the PRZR Safety valves ensures that RCS pressure is limited to less than 110% of design pressure for ALL anticipated transients with the EXCEPTION of an RCP locked rotor.

C. Fifteen (15) minutes; operability of the PRZR Safety valves ensures that RCS pressure is limited to less than 120% of the Safety Limit for RCS pressure for ALL anticipated transients with the EXCEPTION of an RCP locked rotor.

D. Fifteen (15) minutes; operability of the PRZR Safety valves ensures that RCS pressure is limited to less than 110% of design pressure for ALL anticipated transients with the EXCEPTION of an RCP locked rotor.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the wording of the basis is close to the actual wording. Locked rotor events may cause pressure to go to 120% of design pressure. Safety limits are 110% of design pressure

B. Correct. 6 hours because both valves are inoperable. If one valve was inoperable, 15 minutes would be available to correct the problem prior to shutting down

C. Incorrect. 15 minutes is the required TS action time for one safety valve inoperable. Additionally second half wrong because a locked rotor will reach 120% of design, not safety limit

D. Incorrect. Time is incorrect because both valves are affected. Second half is correct

Technical Reference(s): TS 3.4.10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R1901C, Obj 1.13 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 004 2.4.9 | |
| | Importance Rating | | 4.2 |

Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: SRO Question # 89

Initial conditions:

- RCS temperature is 210°.
- The pressurizer is solid.
- PRZR pressure is approximately 350 psig.
- Charging flow is approximately 75 gpm.
- Charging and Letdown are in MANUAL control.

Current conditions:

- RCS pressure is lowering rapidly.
- RCS temperature is stable.
- Charging and Letdown flow are stable in Manual control.
- PRZR level remains off-scale high.

Which ONE of the following describes the procedure entry and initial action that will be required to mitigate this event?

A. AP-RCS.1, Reactor Coolant Leak; Isolate Letdown and control Charging in attempt to maintain PRZR pressure constant.

B. AP-RCS.1, Reactor Coolant Leak; Isolate Letdown and start one SI Pump to prevent a loss of RCS subcooling and maintain RCS inventory.

C. AP-RCS.4, Shutdown LOCA; Isolate Letdown and control Charging in attempt to maintain PRZR pressure constant.

D. AP-RCS.4, Shutdown LOCA; Isolate Letdown and start one SI Pump to prevent a loss of RCS subcooling and maintain RCS inventory.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. This procedure would be entered from higher modes. Plausible because a LOCA is in progress and actions are correct.

B. Incorrect. This procedure would be entered from higher modes. Plausible because a LOCA is in progress

C. Correct. The crew would perform these actions in accordance with the guidance in AP-RCS.4

D. Incorrect. SI pump operation would not be required as an initial action. Alot more inventory would be lost prior to that action

Technical Reference(s): AP-RCS.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RAP18C, Obj 1.02, 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 006 A2.11 | |
| | Importance Rating | | 4.4 |

Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture of ECCS header

Proposed Question: SRO Question # 90

Plant conditions:

- The unit has tripped due to a LOCA.
- RCS pressure - 1450 psig and stable.
- RVLIS Water Level - 65% and stable.
- Containment pressure - 8.5 psig and slowly rising.
- CETs - 810°F and slowly rising.
- RWST level - 70% and lowering.
- Containment sump 78 inches.
- No RCPs are running.
- Containment radiation monitors are in alarm.
- All required ECCS systems are running.
- E-0, Reactor Trip or Safety Injection, is being performed.

1. Which ONE of the following describes the impact on the SI system if a rupture were to occur on the inlet weld to **MOV-878A**, SI PUMP 'A' DISCHARGE ISOL MOV TO LOOP B HOT LEG?

AND

2. Which procedure flowpath will be implemented upon transition from E-0?

A.
1. SI Line to RCS Loop "B" - SI flow on FI-924 and SI pressure on PI-922 would remain stable.
2. E-1 to FR-C.2

B.
1. SI Line to RCS Loop "B" - SI flow on FI-924 would rise and SI pressure on PI-922
2. FR-C.2 to E-0

C.
1. SI Line to RCS Loop "B" - SI flow on FI-924 and SI pressure on PI-922 would remain stable.

D.

2. FR-C.2 to E-0

1. SI Line to RCS Loop "B" - SI flow on FI-924 would rise and SI pressure on PI-922 would lower.

2. E-1 to FR-C.2

Proposed Answer: D

Explanation (Optional):

A. 1. MOV-878A is a normally closed valve that gets SI flow from the "A" SI pump and delivers it to the "B" Hot leg. With the leak on the upstream side where MOV-878B also taps in will result in high flows and lowering pressure on the SI line to the "B" RCS loop.
2. Transition out of E-0 to E-1 prior to start of CSFST monitoring. Once in E-1 CSFST monitoring starts, then go to FR-C.2 then back to E-1, ES-1.2 and ES-1.3. ECA-1.1 will be entered if the leak can't be stopped. Any transition to ECA-1.1 will only happen after ES-1.3 is entered

B. 1. This would be correct if MOV-878A was on a common line that went to the "A" RCS loop but it does not, it is on a common line that goes to the "B" RCS loop.
2. Not monitoring CSFST in E-0 yet, so transition would only happen when in E-1.

C. 1. MOV-878A is a normally closed valve that gets SI flow from the "A" SI pump and delivers it to the "B" Hot leg. With the leak on the upstream side where MOV-878B also taps in will result in high flows and lowering pressure on the SI line to the "B" RCS loop.
2. Not monitoring CSFST in E-0 yet, so transition would only happen when in E-1.

D. 1. MOV-878A is a normally closed valve that gets SI flow from the "A" SI pump and delivers it to the "B" Hot leg. With the leak on the upstream side where MOV-878B also taps in will result in high flows and lowering pressure on the SI line to the "B" RCS loop.
2. Transition out of E-0 to E-1 prior to start of CSFST monitoring. Once in E-1 CSFST monitoring starts, then go to FR-C.2

Technical Reference(s): A-503.1
33013, Sheet 1 and 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R2701C Obj 1.07 (As available)

Question Source: Bank # WTSI 66045

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Ginna 2008

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 2 |
| | K/A # | 068 | 2.1.32 |
| | Importance Rating | | 2.6 |

Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: SRO Question # 91

Plant conditions:

- A Liquid Waste Release has been in progress for 2 hours.
- Reactor power has been reduced from 60% to 49% in the last 60 minutes due to a Circulating Water Pump vibration problem.
- "A" Circulating Water Pump is being removed from service in accordance with T-8A, Startup and Shutdown of Circulating Water Pumps A and B.

Based upon these conditions, which ONE of the following describes the action(s) required?

A. Notify Chemistry to update release rate calculations or stop the release

B. Notify Chemistry to sample the RCS for Iodine and Gross Activity

C. Notify Chemistry to sample the RCS for Iodine and Gross Activity AND notify Chemistry to update release rate calculations or stop the release

D. Direct that the liquid Waste release flow rate be throttled to within the capacity of 1 Circulating Water Pump, and refer to the ODCM

Proposed Answer: A

Explanation (Optional):

A. Correct. If Circ Water Flow Rate is changed, Chemistry must recalculate release rate

B. Incorrect. Power changes >15% in 1 hour require sample

C. Incorrect. Power changes >15% in 1 hour require sample

D. Incorrect. Flow rate will be terminated, not throttled

Technical Reference(s): T-8A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 66157

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Ginna NRC

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 2 |
| | K/A # | 011 2.4.21 | |
| | Importance Rating | | 4.6 |

Emergency Procedures / Plan: 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.

Proposed Question: SRO Question # 92

Plant conditions:

- A Steam Generator Tube Rupture is in progress.
- Complications were observed while performing E-3, Steam Generator Tube Rupture.
- The crew is performing ECA-3.2, SGTR with Loss of Reactor Coolant - Saturated Recovery Desired.
- SI Pumps 'A' and 'C' are running
- RCS subcooling is 0°F.
- RCP 'B' is running.
- RCS fluid fraction is 55%.
- PRZR level is off-scale low.

Which ONE of the following describes the condition of the INVENTORY CSF Status Tree, and the action that will be required?

INVENTORY CSF Status Tree is...

A. GREEN because Safety Injection Pumps are operating. If the Status Tree turns yellow, the CRS would remain in ECA-3.2.

B. YELLOW because Pressurizer level is low. The CRS may perform FR-I.2, Response to Low Pressurizer Level, at his discretion.

C. YELLOW due to voiding in Reactor Vessel; the CRS may perform FR-I.3, Response to Voids in the Reactor Vessel, at his discretion.

D. YELLOW due to voiding in the Reactor Vessel; the CRS will NOT perform actions contained in FR-I.3 because they conflict with the ECA procedure in use.

Proposed Answer: A

Explanation (Optional):

A. Correct. Yellow Path for Inventory will not exist if SI is in service. If Yellow path existed, then it would not be implemented due to conflict with ECA-3.2

B. Incorrect. Yellow Path does not exist with SI pumps running, and action would be normal for yellow conditions, but not for current conditions

C. Incorrect. Plausible because there is a low water density, but with RCP running, FR-I.1 cannot be reached.

D. Incorrect. Plausible because there is a low water density, but with RCP running, FR-I.1 cannot be reached. However, actions are correct for this condition

Technical Reference(s): FR-I.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RFRI3C Obj 2.01, 1.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 2 |
| | K/A # | 001 A2.19 | |
| | Importance Rating | | 4.0 |

Ability to (a) predict the impacts of the following malfunction or operations on the CRDS- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Axial flux distribution

Proposed Question: SRO Question # 93

Plant conditions:

- The plant is operating at 86% power following a downpower due to a Main Condenser leak.
- Control Rod insertion was primarily used.
- Bank D control rods are currently just above the Rod Insertion Limit.

Which ONE of the following describes (1) the direction that Axial Flux Difference (AFD) will trend, and (2) the action required if AFD exceeds the limit specified in the Core Operating Limits Report (COLR)?

AFD will trend in the...

A. POSITIVE direction; restore AFD within 30 minutes or be in Mode 3 within 6 hours.

B. POSITIVE direction: reduce reactor power to <50% within 30 minutes of exceeding the limit.

C. NEGATIVE direction; restore AFD within 30 minutes or be in Mode 3 within 6 hours.

D. NEGATIVE direction; reduce reactor power to <50% within 30 minutes of exceeding the limit.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. AFD will trend in the negative direction because flux will trend toward the bottom of the core with rod insertion. Action is plausible because 30 minutes is part of the action statement, and the 6 hours represents normal shutdown to Mode 3 action statements

B. Incorrect. Plausible as in A, but action to restore is correct. Applicant may confuse positive versus negative if they misunderstand the calculation for AFD

C. Incorrect. Right direction but wrong action. Plausible as described in Option A

D. Correct.

Technical Reference(s): TS 3.2.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: Needed (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Using a procedure number was redundant so I eliminated. This is SRO based upon the TS action

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 1 |
| | K/A # | G1 2.1.25 | |
| | Importance Rating | | 4.2 |

Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: SRO Question # 94

Plant conditions:

- The plant was operating at 100% power when a reactor trip occurred on low pressurizer pressure.
- "B" S/G Tube Rupture was diagnosed, and E-3, Steam Generator Tube Rupture, was entered.
- RCS Cooldown and Depressurization is complete.

Plant conditions:

- SG "B" level is 32% and rising.
- SG "A" level is 52% and stable.
- PRZR level is 37% and lowering.

Which ONE of the following describes the required operator action IAW E-3, and which ONE of the following procedures will subsequently be used for the ruptured SG Cooldown if radioactive release and contamination must be minimized?

(Reference Provided)

A. Depressurize RCS; ES-3.1, Post SGTR Cooldown Using Backfill

B. Depressurize RCS; ES-3.2, Post SGTR Cooldown Using Blowdown

C. Energize PRZR Heaters; ES-3.2, Post SGTR Cooldown Using Blowdown

D. Energize PRZR Heaters; ES-3.1, Post SGTR Cooldown Using Backfill

Proposed Answer: A

Explanation (Optional):

A. Correct. Table shows with PRZR level between 20% and 50% and SG level rising, RCS must be depressurized.

B. Incorrect. Plausible because action is correct, but preferred procedure for conditions indicated is ES-3.1

C. Incorrect. Plausible because this action is in the column next to the required action. If SG level were lowering, this action would be taken. ES-3.2 would be used if there was a reason NOT to use ES-3.1 for SG cooldown under these conditions.

D. Incorrect. Plausible because the correct procedure is indicated. Action would be correct if SG level was lowering.

Technical Reference(s): E-3, Rev 46, Step 36 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: E-3, Step 36

Learning Objective: REP03C, Obj 2.01 (As available)

Question Source: Bank #

Modified Bank # WTSI 66155 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Modified from 2007 Ginna NRC exam. Changed levels and direction. Replaced first part of distracter A and B and made conditions so that there is a new correct answer.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 2 |
| | K/A # | G2 2.2.44 | |
| | Importance Rating | | 4.4 |

Equipment Control: 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.

Proposed Question: SRO Question # 95

Plant conditions:

- A reactor trip and safety injection have occurred.
- All equipment is operating as designed.
- The crew is performing diagnostic actions of E-0, Reactor Trip Or Safety Injection.
- Containment pressure is 19 psig and LOWERING.
- RCS pressure is 1250 psig and STABLE.
- RCS subcooling margin is 46°F and STABLE.
- RWST level is 77% and dropping slowly
- PRZR level is 4% and RISING.
- All AFW pumps are running with 400 gpm total flow.
- BOTH RCP's have been secured.
- The Shift Manager intends to restart one RCP as soon as is practical.

Which ONE of the following procedures will be in use when the crew restarts a reactor coolant pump?

A. ES-1.1, SI Termination

B. ES-1.2, Post-LOCA Cooldown And Depressurization

C. ES-1.3, Transfer to Cold Leg Recirculation

D. E-1, Loss Of Reactor Or Secondary Coolant

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Conditions do not exist to transition to ES-1.1. RCS pressure must be greater than 1650 psig. Plausible because conditions are stable and parameters are

recovering.

B. Correct. From E-0, crew will transition to E-1. In E-1, transition will be made to ES-1.2, where an RCP may be started for RCS cooldown.

C. Incorrect. Plausible because a LOCA is in progress and RWST level is lowering. However, RWST is not at a level where transition to ES-1.3 is imminent, so this procedure will not be the next one entered.

D. Incorrect. Plausible because E-1 is the next procedure entered. However, E-1 stabilizes conditions and verifies equipment operation prior to transition to another procedure. Actions such as starting RCPs are performed in the subsequent procedure; in this case, ES-1.2

Technical Reference(s): ES-1.2, Rev 33, Step 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RES12C, Obj 2.01 (As available)

Question Source: Bank # WTSI 55134

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Beaver Valley Unit 2 2005

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 3 |
| | K/A # | G3 2.3.5 | |
| | Importance Rating | | 2.9 |

Radiation Control: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: SRO Question # 96

Plant conditions:

- The plant is in Mode 6.
- Core alterations are in progress.
- A Containment Purge is being prepared in accordance with S-23.2.2, Containment Purge Procedure.

Which ONE of the following describes (1) an acceptable ventilation alignment, and (2) the radiation monitors required to be operable for these conditions?

A. (1) 2 Purge Supply Fans running; 1 Purge Exhaust Fan running; Containment
(2) R-11 and R-12 MUST be operable for the release to proceed

B. (1) 2 Purge Supply Fans running; 1 Purge Exhaust Fan running; 1
(2) R-12A MAY be used to satisfy the requirements for the release

C. (1) 1 Purge Supply Fan running; 2 Purge Exhaust Fans running; Containment
(2) R-11 and R-12 MUST be operable for the release to proceed

D. (1) 1 Purge Supply Fan running; 2 Purge Exhaust Fans running; 1
(2) R-12A MAY be used to satisfy the requirements for the release

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Not acceptable to have 2 supply and 1 exhaust fan. Negative pressure in containment is required. R-11 and R-12 are not required in Mode 6, the applicability is for modes 1-4 and procedure allows use of R-12A for this evolution in Mode 5 or 6

B. Incorrect. Not acceptable to have 2 supply and 1 exhaust fan. Negative pressure in containment is required. Correct application of radiation monitor use

C. Correct. Because refueling is in progress, R-11 and R-12 must be operable

D. Incorrect. R-12A cannot satisfy requirements for release with refueling in progress

Technical Reference(s): S-23.2.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # WTSI 66188 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

Modified by changing 2nd part

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 4 |
| | K/A # | G4 | 2.4.41 |
| | Importance Rating | | 4.6 |

Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications.

Proposed Question: SRO Question # 97

Initial Conditions:

- A controlled shutdown was being performed due to a 200 GPD primary-to-secondary leak on "B" steam generator.
- During the shutdown, SG tube leakage increased to approximately 150 GPM.
- The reactor did NOT automatically trip when required.
- Manual reactor trip from the MCB was successful.
- The CO then reported all steam generator pressures at 850 psig and lowering rapidly.
- The crew manually initiated Safety Injection and Main Steam Isolation signals.
- All ESF systems functioned properly.

Current Conditions:

- Containment pressure is 0.6 psig and stable.
- "B" steam generator pressure continues to lower rapidly.

Which ONE of the following describes the HIGHEST emergency classification for this event?

(Reference Provided)

A. Alert (1.1.1)

B. Alert (3.1.2)

C. Site Area Emergency (1.1.2)

D. Site Area Emergency (3.2.2)

Proposed Answer: D

Explanation (Optional):

A. Incorrect because the classification is not high enough, although conditions are met for this call because the reactor did not automatically trip

B. Incorrect because the classification is not high enough, although conditions are met for this call. Leakage is higher and there is an uncontrolled depressurization of SG secondary side

C. Incorrect because the conditions are not me for this classification. There is no Subcriticality red path for this event

D. Correct

Technical Reference(s): EPIP-1.0, EALs (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EPIP-1.0, EAL Tables

Learning Objective: Needed (As available)

Question Source: Bank # WTSI 63633

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Harris 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 3 |
| | K/A # | G3 2.3.4 | |
| | Importance Rating | | 3.7 |

Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: SRO Question # 98

Plant conditions:

- A General Emergency is in progress.
- A worker is critically injured and unconscious in the RHR pit.
- Extremely high radiation levels exist in the area.
- The Duty RP Tech is determining expected dose for two volunteers that will be assigned Search and Rescue responsibilities in accordance with EPIP 1.8, Search and Rescue.

Which ONE of the following describes the maximum dose guideline and highest approval required to allow the dose in accordance with EPIP 2-8, Voluntary Acceptance of Emergency Radiation Exposure?

A. 25 Rem under all circumstances

B. >25 Rem only if the volunteers are fully aware of the risks involved

C. 10 Rem under all circumstances

D. >10 Rem but less than or equal to 25 Rem only if the volunteers are fully aware of the risks involved

Proposed Answer: B

Explanation (Optional):

A. Incorrect. 25 Rem is the guideline but that dose may be exceeded by volunteers that have been briefed on the risks involved.

B. Correct.

C. Incorrect, This value is for saving plant equipment/protecting valuable property

D. Incorrect. Same as C but the remainder of this option is logical and consistent with B

Technical Reference(s): EPIP-2.8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RSCO2C 17.00 (As available)

Question Source: Bank #

Modified Bank # WTSI 66461 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 4 |
| | K/A # | G4 2.4.38 | |
| | Importance Rating | | 4.4 |

Emergency Procedures / Plan: Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

Proposed Question: SRO Question # 99

Which ONE of the following Emergency Coordinator responsibilities may be delegated to another individual?

A. Classifying and declaring emergencies and requesting the formation of emergency teams.

B. Directing operations of emergency response organizations and providing Off-Site Protective Action Recommendations.

C. Initiating the implementation of on-site protective actions and directing operations of emergency response organizations.

D. Contacting Off-Site authorities by telephone in accordance with EPIP-1.5.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Classification is a non-delegable responsibility of the EC. Plausible because once the ERO is staffed, the TSC provides a significant amount of input to the EC.

B. Incorrect. PARs are responsibility of the EC. Plausible because once the ERO is staffed, the TSC provides a significant amount of input to the EC.

C. Incorrect. On-site actions are directed by EC. Plausible because on-site organizations such as the OSC are directed by their own coordinators, but protective actions are responsibility of the EC.

D. Correct.

Technical Reference(s): EPIP-1.5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 55219

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Callaway 2005

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 1

Conditions and limitations in the facility license

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 1 |
| | K/A # | G1 2.1.42 | |
| | Importance Rating | | 3.4 |

Conduct of Operations: Knowledge of new and spent fuel movement procedures.

Proposed Question: SRO Question # 100

Plant conditions:

- The plant is in Mode 6.
- Refueling is in progress.
- RF-301, Refueling Operations (Offload, Shuffle, Reload) is being performed for a core offload.
- A fuel assembly is in transit from its position in the core to the Upender.
- Refueling Cavity level begins to lower rapidly.
- Sump A level is rising rapidly.
- The Refueling SRO evacuates Containment and notifies the Control Room.

Which ONE of the following describes additional action that will be required in accordance with RF-601, Fuel Handling Accident Instructions?

A. Return the fuel assembly to its position in the core; return the Fuel Transfer Cart to the Spent Fuel Pit.

B. Return the fuel assembly to its position in the core; Ensure the transfer cart is in the Refueling Cavity.

C. Place the fuel assembly in the emergency location in the fuel transfer slot and lower to the bottom of the slot area; Ensure the transfer cart is in the Refueling Cavity.

D. Place the fuel assembly in the emergency location in the fuel transfer slot and lower to the bottom of the slot area; return the Fuel Transfer Cart to the Spent Fuel Pit.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Would place fuel assembly here if it was over its core position, but not in transit to the upender.

B. Incorrect. Would place fuel assembly here if it was over its core position, but not in transit to the upender. Also, Transfer cart belongs in SFP so that the transfer tube gate valve can be closed.

C. Incorrect. Correct location for fuel assembly but Transfer cart belongs in SFP so that the transfer tube gate valve can be closed.

D. Correct.

Technical Reference(s): RF-601, Rev 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: N/A (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 7

Fuel handling facilities and procedures.

Comments: