Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AK3.05
	Importance Rating	3.2

Partial or Complete Loss of Forced Core Flow Circulation:

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reduced loop operating requirements: Plant-Specific

Proposed Question: #1

The Plant was operating at full power when the following occurs:

- A lockout condition on the 10300 bus (4160 VAC bus) is generated
- Reactor Recirculation Pump (RWR) 'B' trips on overcurrent

Based on the above conditions, which one of the following is **(1)** the correct Operator action to be performed and **(2)** the reason for that action?

	(1)	(2)
A.	Establish Single Loop Operation (SLO)	to stop the idle RWR Pump 'B' impeller rotation
B.	Establish Single Loop Operation (SLO)	to activate SLO Rod Block and Scram setpoints
C.	Insert a Manual Scram	due to no Forced Circulation through the Core
D.	Insert a Manual Scram	to preclude a Cold Water Accident reactivity addition

Proposed Answer: C

Explanation: With the Plant operating at Full Power, the Mode Switch is in RUN. A lockout condition on the 10300 bus results in that bus de-energizing and remains de-energized until repaired. When the 10300 bus de-energizes, two RWR MG Set ‘A’ Lube Oil Pumps 02-184P-2A1 and 2A3 trip. This results in a trip of the ‘A’ RWR pump on low Lube Oil pressure (see AOP-16, Loss 10300 Bus). The second bullet in the stem states that RWR ‘B’ trips on overcurrent. This trip results in no RWR pumps running. AOP-8 (Loss or Reduction of Reactor Coolant Flow) is entered. Immediate Actions and Override statements in AOP-8 direct a Manual Scram be inserted if no RWR pumps are running with the Mode Switch in RUN.

A. Incorrect: SLO would be established to stop an idle RWR if one RWR pump was running
B. Incorrect: SLO would be established to activate SLO setpoints if one RWR pump was running
D. Incorrect: Scram is required however; with no RWR pumps available to be started a CWA cannot occur

Technical Reference(s): AOP-8, AOP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H: 1.10, 1.14, 1.15

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AA2.02
	Importance Rating	4.2

Partial or Complete Loss of AC

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Reactor power / pressure / and level

Proposed Question: #2

The Plant has experienced a station blackout and a failure of the UPS inverter to transfer to DC.

Which one of the following combinations correctly lists the indicators (meters, lights, etc) the Crew can use to determine Reactor Power and Level?

	Reactor Power	Reactor Level
A.	IRM Downscale lights on Panel 09-5	Fuel Zone Level Indicator 02-3LI-92 on Panel 09-4
B.	IRM \ APRM Recorder on Panel 09-5	Wide Range Level Recorder 06LR-97 on Panel 09-5
C.	IRM Downscale lights on Panel 09-5	Wide Range Level Recorder 06LR-97 on Panel 09-5
D.	IRM \ APRM Recorder on Panel 09-5	Fuel Zone Level Indicator 02-3LI-92 on Panel 09-4

Proposed Answer: A

Explanation: Conditions given in Stem are a loss of all AC power (Station Blackout) coupled with a loss of the Uninterruptible Power Supply (UPS). The only correct combination of indicators available to monitor Rx power and level under this condition is A. See below:

AOP-21: **IF** UPS Inverter failed to transfer to DC, **THEN** perform the following:

E.3.1 Monitor RPV water level on the following indicators:

- RX WTR LVL 02-3LI-85A (09-5)
- RX WTR LVL B 06LI-94B (09-5)
- RX WTR LVL C 06LI-94C (09-5)
- Rx WTR LVL FUEL ZONE 02-3LI-91 (09-3)
- **RX WTR LVL FUEL ZONE 02-3LI-92 (09-4)**

Verify reactor power is lowering using the following indications:

- SRM meters and **IRM downscale indicators** at panel 09-5

Technical Reference(s): AOP-21, AOP-49

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71E 1.14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

DK 4/11/12: Deleted Pressure column with NRC concurrence

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AK1.05
	Importance Rating	3.3

Partial or Complete Loss of DC Pwr

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Loss of breaker protection

Proposed Question: #3

With the Plant operating at 92% power and RHR Loop 'A' in Torus Cooling, the following occurs:

At time 0:00 minutes, a complete loss of Station Battery 'A' occurs and the following Annunciators are received:

- 09-8-1-20 125VDC BATT A VOLT LO
- 09-3-1-05 RHR PMP 10P-3A TRIP OR CNTRL PWR LOSS

At time 0:05 minutes, the following is noted:

- 10MOV-13A (Torus Suction Valve) strokes closed

Based on these conditions, which one of the following states the implication (design or Operator action) for the above?

RHR Pump 10P-3A …

A. automatically tripped when Station Battery 'A' was lost at time 0:00 minutes

B. automatically tripped when Torus suction valve 10MOV-13A repositioned at time 0:05 minutes

C. continues to run and must be tripped locally at the breaker

D. continues to run and must be tripped from the Control Room

Proposed Answer: C

Explanation: RHR pumps receive a trip signal on a loss of suction path. 10MOV-13A is the suction valve for RHR Pump 3A. A loss of Station Battery A is entry to AOP-45 (Loss of DC Power System A). AOP-45 states: **CAUTION** Loss of breaker control power disables all protective breaker trips. Therefore, with the suction valve shut, RHR Pump 3A should have automatically tripped, but it will not trip due to the loss of DC control power. With no DC control power, there is no power to energize the Trip Coil. The control switch in the Control Room also will have no effect. The only method available for an Operator to secure RHR Pump 3A, is to locally trip the power supply breaker.

Technical Reference(s): OP-13, AOP-45, ARP-09-3-1-05

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

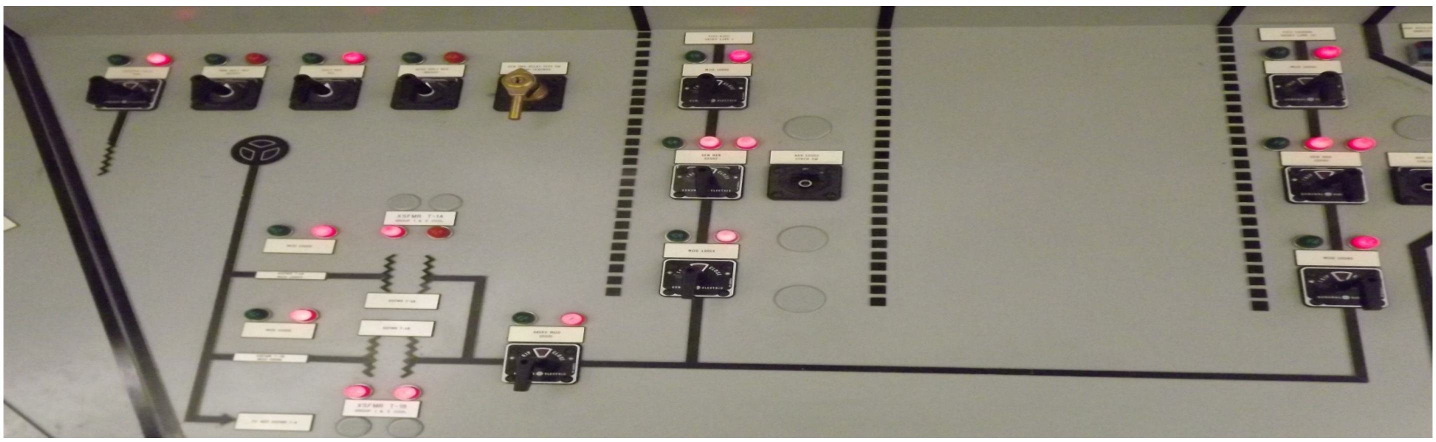
Comments:

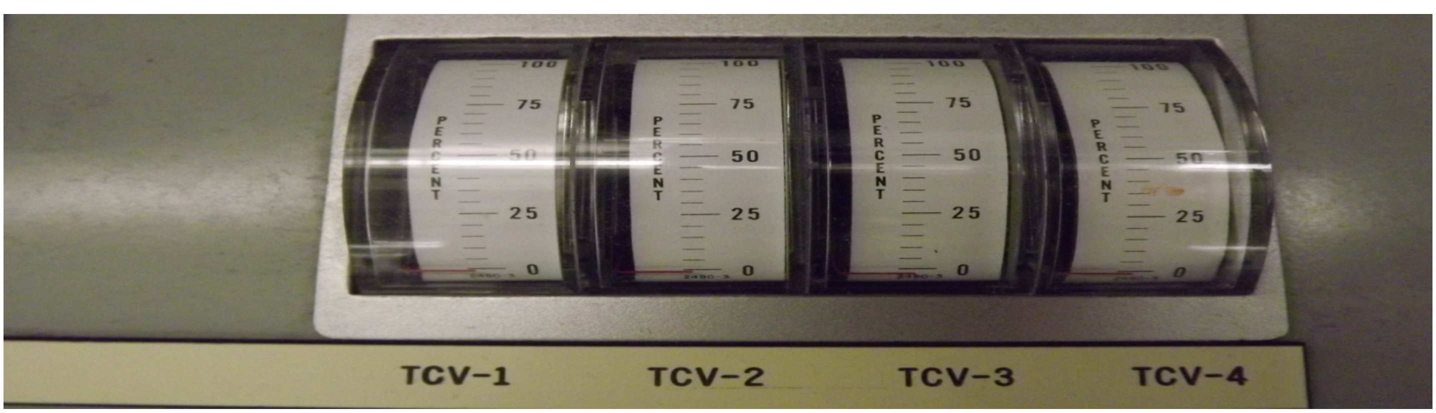
DK 4/11/12 - Reformatted for clarity based on NRC review.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 2.1.31
	Importance Rating	4.6

Main Turbine Generator Trip
Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.
Proposed Question: #4

The Plant was operating at 22% power when an automatic Rx scram occurred. The following pictures show present Plant conditions:





Based on <u>these indications</u>, which one of the following correctly states the status of the Main Generator and the Main Turbine?

	Main Generator	Main Turbine
A.	connected to Grid	**NOT** tripped
B.	connected to Grid	tripped
C.	**NOT** connected to Grid	tripped
D.	**NOT** connected to Grid	**NOT** tripped

Proposed Answer: B

Explanation: A Rx scram results on a Main Generator trip followed by a Turbine trip (not be confused with the concept that a Turbine trip <29% does NOT result in a Rx scram). When the Turbine trips the following occurs:

Turbine Stop Valves, Intermediate Stop Valves, Control Valves, and Intercept Valves close.
Main Generator output breakers 10042 and 10052 open initiating a fast transfer of house loads.

The top two pictures show the Turbine Stop and Control Valve Positions in percent open. 0% indicates full closed; as expected on a Turbine trip. The third picture shows an abnormal alignment for the 10042 and 10052 breakers; red light on, green light off indicates the breakers are closed and the Main Generator is still connected to the 345KV Grid.

Technical Reference(s): OP-9, AOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94a 1.09

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments: moved picture of electric plant to before pictures of turbine. Validators are not believing what electric plant pic is showing. 7/3/DK

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 AA2.01
	Importance Rating	4.5

SCRAM

Ability to determine and/or interpret the following as they apply to SCRAM: Reactor power

Proposed Question: #5

The Plant was operating at full power with HPCI tagged out of service when a spurious Group 1 Primary Containment Isolation signal occurred.

The following conditions exist:

- Rx Mode Switch is in Shutdown
- All Control Rods are NOT full in
- Four SRVs are open
- Rx Pressure is 1140 psig and steady
- RPV level is 180 inches and steady
- RCIC is not running

Which one of the following ranges contains the approximate value of Reactor Power?

A. 5-10%

B. 15-20%

C. 30-35%

D. 55-60%

Proposed Answer: C

Explanation: A Group 1 isolation signal causes the MSIVs to close. When the MSIVs are not full open, a Rx Scram is generated. Conditions given indicate an equilibrium condition exists since RPV pressure and level are steady (i.e. the power production in the core equals the steam flow thru the SRVs.) Each SRV will pass approximately 900,000 lbm\hr at 1145 psig. Therefore 4 SRVs = 3.6 x 10^6 lbm\hr. Total Steam Flow at 100% power is approximately 10.97 x 10^6 lbm\hr; therefore each SRV can pass approximately 8.2% Rx power. Four SRVs multiplied by 8.2 is ~ = 32%. With HPCI and RCIC not running, there are no other steam loads since the MSIVs are shut. Choice D is plausible should Candidate believe MSIVs are open <u>and</u> the Turbine Bypass Valves were open (TBV capacity is 25% Rx power [25 + 32]). Choice A and B are plausible should Candidate not determine correct SRV capacity and relate it to Total Steam Flow.

Technical Reference(s): FSAR Table 4.4-1 and Chapter 16.9.2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J, 1.05 and SDLP-29, 1.02, 1.05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

DK 4/11/12 - Changed choice values from single number to small range, based on NRC review.

DK/TRH 4/16/12 - Adjusted answer choice ranges to be wider and slightly reworded question to reference ranges.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AA1.01
	Importance Rating	3.8

Control Room Abandonment

Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: RPS

Proposed Question: #6

The Plant was operating at 78% power with the following RPS electrical power alignment:

- RPS ‘A’: powered from its Normal Power Supply
- RPS ‘B’: powered from its Alternate Power Supply

The following takes place:

- A significant fire in the Relay Room occurs.
- Smoke from the fire requires the Control Room to be abandoned.
- AOP-43 (Plant Shutdown From Outside The Control Room) is entered.

All actions by the ATC (At the Controls) Operator to scram the Reactor from the Control Room fail.

- No Control Rods move.
- RPS Group ‘A’ and ‘B’ white lights are ON.

Which one of the following states how the Reactor is to be shutdown?

(1) <u>RPS ‘A’</u>
(2) <u>RPS ‘B’</u>

A. **(1)** Place RPS MG Set ‘A’ GENERATOR OUTPUT breaker in OFF
(2) Place RPS MG Set ‘B’ GENERATOR OUTPUT breaker in OFF

B. **(1)** Place RPS MG Set ‘A’ GENERATOR OUTPUT breaker in OFF
(2) Open breaker 71MCC-262-OB2 71-05-6B REACTOR PROTECTION BUS B DISTRIBUTION PANEL.

C. **(1)** Open breaker 71MCC-252-OC1 71-05-6A REACTOR PROTECTION BUS A DISTRIBUTION PANEL.
(2) Place RPS MG Set ‘B’ GENERATOR OUTPUT breaker in OFF

D. **(1)** Open breaker 71MCC-252-OC1 71-05-6A REACTOR PROTECTION BUS A DISTRIBUTION PANEL.
(2) Open breaker 71MCC-262-OB2 71-05-6B REACTOR PROTECTION BUS B DISTRIBUTION PANEL.

Proposed Answer: B

Explanation: RPS ‘A’ being supplied by its Normal power supply implies it is powered from its Motor Generator set. RPS ‘B’ is being powered from its Alternate power supply (71-05-6B). Per AOP-43 with the white RPS Group lights on and Rx power >2.5%, direction is given to de-energize the RPS busses by Choice B only. Choice A would result in a half-scram on ‘A’ side only. Choice C would result in no scram and Choice D would result in a half-scram on ‘B’ side only.

Technical Reference(s): AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.5, 1.11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AK1.01
	Importance Rating	3.5

Partial or Complete Loss of CCW

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations

Proposed Question: #7

The Plant was operating at 85% power with Reactor Building Closed Loop Cooling (RBCLC) pump 15P-2B danger tagged out of service.

The following occurs:

- The L13 (600VAC) bus de-energizes

Which one of the follows states **(1)** the correct Operator action and **(2)** a reason for the action?

	(1) Operator action	**(2) reason for action**
A.	Perform a Rapid Power reduction per RAP 7.3.16	lower Reactor Water Cleanup heat load
B.	Scram the Reactor per AOP-1	the Main Steam Isolation Valves must be shut
C.	Perform a Rapid Power reduction per RAP 7.3.16	lower Main Turbine Generator heat load
D.	Scram the Reactor per AOP-1	the Reactor Recirculation Pumps must be tripped

Proposed Answer: D

Explanation: L13 is the power supply to RBCLC pumps ‘A’ and ‘C’. With RBCLC pump ‘B’ already out of service, this results in no RBCLC pumps running. This is an entry to AOP-11 (Loss of RBCLC). The Immediate Actions of AOP-11 require the Rx scrammed and RWR pumps tripped in that sequential order. RWR pumps are tripped due to loss of cooling to their MG Fluid Drive Oil Coolers. A rapid power reduction would be appropriate under a loss of Turbine Building Closed Loop (TBCLC) condition. A Rapid Power reduction would lower RWCU heat load and lower Main Generator heat load (plausible but not correct for these conditions). Closing MSIVs would be appropriate if the Candidate believed an emergency shutdown of the Main Turbine must be performed.

Technical Reference(s): AOP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15 1.15

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AA1.02
	Importance Rating	3.3

Partial or Complete Loss of Inst. Air

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system valves: Plant-Specific

Proposed Question: #8

The Plant is operating at full power when a report from the field is received concerning an unisolable leak from the Instrument Air (IA) header.

IA header pressure is 100 psig lowering slowly.

Which one of the following lists which IA system valve would close first and total number of IA compressors that would be running in an attempt to mitigate the lowering pressure?

	First IA valve to close	IA compressors running
A.	39FCV-110 (Service Air Header Auto Isolation Valve)	2
B.	39AOV-111 (Breathing Air Header Isolation Valve)	2
C.	39AOV-111 (Breathing Air Header Isolation Valve)	3
D.	39FCV-110 (Service Air Header Auto Isolation Valve)	3

Proposed Answer: D

Explanation: Lowering IA header pressure results in the following automatic actions:

- 107# - first standby IA compressor starts
- 104# - second standby IA compressor starts
- 95# - 39FCV-110 closes
- 85# - 39AOV-111 closes

Therefore at 100 psig, all three IA compressors would be running and 39FCV-110 would close first in an attempt to mitigate the pressure reduction.

Technical Reference(s): AOP-12

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.05, 1.08, 1.10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.03
	Importance Rating	3.5

Loss of Shutdown Cooling

Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water level

Proposed Question: #9

24 hours ago, the Plant entered into a Forced Outage with the following conditions:

- RHR Loop ‘B’ is in a Shutdown Cooling lineup
- Recirculation Pump (RWR) ‘A’ is the only RWR Pump running
- RPV level is 199 inches

The following conditions now exist:

- RWR ‘A’ Drive Motor lube oil pressure: 25 psig steady
- RPV pressure: 80 psig rising slowly

Which one of the following choices is correct concerning RPV level under these conditions?

RPV level is…

A. sufficient, maintain 197-207 inches (green band).

B. insufficient, raise level to 200-234 inches.

C. sufficient, maintain 177-222.5 inches.

D. insufficient, raise level to 235-270 inches.

Proposed Answer: D

Explanation:

CAUTION If RPV water level is less than 234.5 inches with no forced core recirculation, reactor coolant temperature indications could be invalid due to insufficient natural circulation. AOP-30 states:
RPV WATER LEVEL CONTROL
Attempt to maintain RPV water level as follows:

A. **No** RWR pump running: **BETWEEN** 234.5 and 270 inches.
B. RWR pump running: **BETWEEN** 200 and 270 inches.

Technical Reference(s): OP-13D, AOP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.09, 1.15

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AA2.03
	Importance Rating	3.3

Refueling Acc

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Airborne contamination levels

Proposed Question: #10

The Plant is operating at full power with ISFSI cask loading activities in progress when a spent fuel element is dropped.

The following occurs:

- 17RM-456A (Refuel Floor Exhaust Radiation Monitor) spikes high offscale THEN immediately fails downscale.
- 17RM-456B (Refuel Floor Exhaust Radiation Monitor) detects $2x10^4$ cpm (steady value).
- 17RM-452A (Below Refuel Floor Exhaust Radiation Monitor) reads 30 cpm (steady value).
- 17RM-452B (Below Refuel Floor Exhaust Radiation Monitor) reads 200 cpm (steady value).

Which one of the following lists the expected response of the Reactor Building (RB) Ventilation system?

RB Ventilation train(s)…

A. 'A' and 'B' remain in service

B. 'A' and 'B' both isolate

C. 'A' isolates only

D. 'B' isolates only

Proposed Answer: B

Explanation: 17RM-456A and B (Refuel Floor Exhaust Radiation Monitor) setpoints are:

- Downscale/inop – 10cpm
- High – $1x10^3$ cpm (EPIC only)
- High – $5x10^3$ cpm
- Hi-Hi – $1x10^4$ cpm

In the event of high-high radiation on either channel or downscale/inop on both channels, a R.B. Ventilation isolation signal will be generated. The Hi-Hi trip value "locks-in" and must be manually reset. Therefore, even though a momentary Hi-Hi trip on Channel A was generated, followed by a downscale condition, the RB Ventilation system will isolate (trip). The Above and Below Refuel Monitors are independent monitors and either detector can\will isolate RB Ventilation.

Technical Reference(s): OP-51A, ARP-09-3-2-40

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-66A 1.14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments: 7/3/DK changed trip to isolates and below floor values based on val comment

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK1.01
	Importance Rating	4.1

High Drywell Pressure

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Drywell integrity: Plant-Specific

Proposed Question: #11

A loss of coolant accident has occurred and the following parameters are present:

- RPV pressure: 725 psig and lowering
- Drywell pressure: 5 psig and rising
- Torus pressure: 8 psig and rising
- Drywell temperature: 305°F and rising

Based on the above conditions, **(1)** is Drywell spray allowed and **(2)** why \ why not?

	(1) Drywell Spray allowed?	(2) why \ why not?
A.	Yes	Drywell temperature is approaching 309°F
B.	Yes	Torus pressure is above 2.7 psig
C.	No	Torus pressure is NOT above 15 psig
D.	No	Excessive Torus to Drywell dP could result

Proposed Answer: D

Explanation: Drywell Spray Initiation Limit: The Drywell Spray Initiation Limit is defined to be the highest drywell temperature at which initiation of drywell sprays will **NOT** result in an evaporative cooling pressure drop to below:
The high drywell pressure scram setpoint (2.7 psig).

This limit is a function of drywell pressure, and is utilized to **preclude containment failure** or de-inertion following initiation of drywell sprays. With typical DW spray flow rates, **this cooling process results in an immediate rapid and large reduction in the drywell pressure at a rate much faster than can be compensated for by the vacuum relief system**. For low drywell pressures, the high drywell pressure scram setpoint limits the evaporative cooling pressure drop so that the allowable pressure drop increases with increasing drywell pressure. Thus, the first segment reflects an increasing limit with increasing drywell pressure.

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

Proposed references to be provided to applicants during examination: EOP-11 DWSIL curve

Learning Objective: MIT-301.11B 1.01

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 2.4.46
	Importance Rating	4.2

High Reactor Pressure

Ability to verify that the alarms are consistent with the plant conditions.

Proposed Question: #12

The Plant is operating at full power when Annunciator 09-5-1-22 (RPS HI RX PRESS TRIP) alarms.

The following is reported:

- Pressure transmitter 02-3PT-55A senses 1038 psig and steady
- Pressure transmitter 02-3PT-55B senses 1041 psig and steady
- Pressure transmitter 02-3PT-55C senses 1047 psig and steady
- Pressure transmitter 02-3PT-55D senses 1083 psig and steady

Which one of the following states the expected response of the RPS Auto Scram Annunciators, 09-5-1-03 and 09-5-1-04, based on the sensed pressures?

	09-5-1-03 RPS A AUTO SCRAM	**09-5-1-04 RPS B AUTO SCRAM**
A.	In alarm	NOT in alarm
B.	NOT in alarm	In alarm
C.	In alarm	In alarm
D.	NOT in alarm	NOT in alarm

Proposed Answer: B

Explanation: Annunciator 09-5-1-22 (RPS HI RX PRESS TRIP) setpoint is 1062 psig (TS setpoint must be $\leq$1080 psig). Only the 'B' side RPS logic is satisfied with the given values therefore a "half-scram" on the 'B' side will be processed. This results in only the RPS B Annunciator alarming ('A' side would still be off).

Technical Reference(s): Annunciator 09-5-1-22, 09-5-1-03, 9-5-1-04

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.11, 1.12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EK1.02
	Importance Rating	3.5

Suppression Pool High Water Temp

Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Steam condensation

Proposed Question: #13

With the Plant operating at 98% power, the Operating Crew is preparing to perform HPCI Surveillance Test ST-4N.

A Caution in the ST states; “During testing which adds heat to the Torus, Torus water temperature shall not exceed 105°F”.

Which one of the following states the bases for the above ST-4N Caution?

Torus temperatures greater than 105°F…

A. reduces the overall steam condensing capability of the Torus.

B. could result in structural failure should Torus Spray be needed.

C. will result in a rise in Torus water level which could jeopardize the operability of the Safety Relief Valves.

D. creates a localized “hot-spot” of the HPCI turbine exhaust which could lead to unacceptable thermal stress.

Proposed Answer: A

Explanation:

ST-4N Precaution 6.1 states:
6.1.5 During testing which adds heat to the torus, torus water temperature shall not exceed 105°F. **IF** Temperature exceeds 105°F, notify CRS and immediately suspend all testing that adds heat to the torus.
IF temperature exceeds 110°F, a scram is required.

3.6.2.1 Suppression Pool Average Temperature

LCO 3.6.2.1 Suppression pool average temperature shall be:

a. $\leq$ 95°F with THERMAL POWER > 1% RTP and no testing that adds heat to the suppression pool is being performed;
b. $\leq$ 105°F with THERMAL POWER > 1% RTP and testing that adds heat to the suppression pool is being performed; and
c. $\leq$ 110°F with THERMAL POWER < 1% RTP.

C. Suppression pool average temperature > 105°F
AND
THERMAL POWER > 1% RTP
AND
Performing testing that adds heat to the suppression pool.

C.1 Suspend all testing immediately that adds heat to the suppression pool.

BASES: The technical concerns that lead to the development of suppression pool average temperature limits are as follows:

a. Complete steam condensation

Technical Reference(s): ST-4N, TS 3.6.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A 1.17

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 EK2.03
	Importance Rating	3.6

High Drywell Temperature

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Reactor water level indication

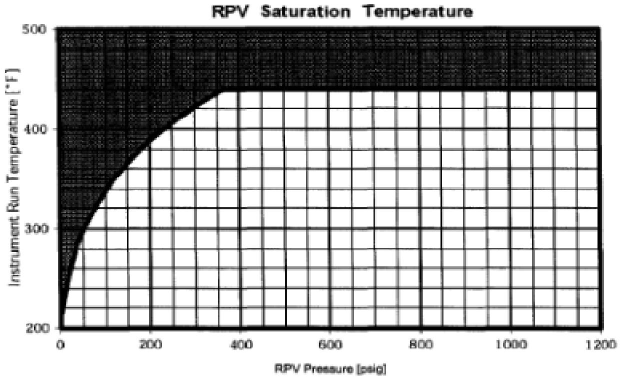
Proposed Question: #14

The Plant is commencing a Shutdown. The Drywell has been de-inerted when a fire occurred in the Drywell.

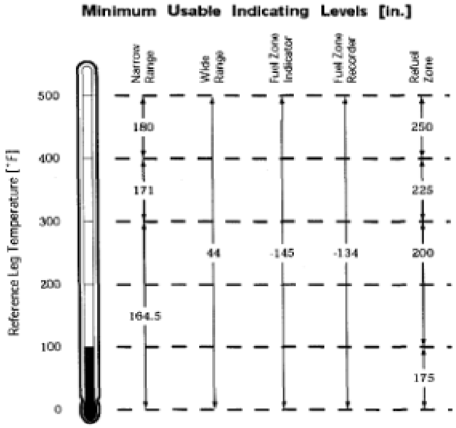
The following conditions are present:

- RPV pressure: 620 psig
- Drywell temp: 320°F
- RPV level:
    1. Narrow Range: 165 inches
    2. Refuel Zone: 180 inches
    3. Wide Range: 160 inches

RPV Saturation Temperature



Minimum Usable Indicating Levels [in.]



Which one of the following choices lists the RPV level instruments that can be used to determine RPV level?

A. 1 only

B. 3 only

C. 1 and 2 only

D. 2 and 3 only

Proposed Answer: B

Explanation: With a DW temp of 320F, the RPV Saturation Temperature curve is in the “good” region. This would imply that all instruments can be used; however bullet #2 of the Curve states: An RPV water level instrument may be used to determine RPV water level only if the instrument reads above its Minimum Usable Indicating Level. The only instrument that meets this criterion is the Wide Range level indicator. All values given could be misinterpreted as being good \ plausible values.

	Minimum value	Indicating value
Wide Range	44	160
Narrow Range	171	165
Refuel Zone	225	180

Technical Reference(s): EOP-11

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank

Question History: 2012 LOI-12-1 Question #68 (see attached Parent)

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: DK 4/23: embedded curve and min values in question. Deleted EOP-11 from Reference provided.

Modified Bank Parent Question

The Plant was operating at full power when a significant event occurred.

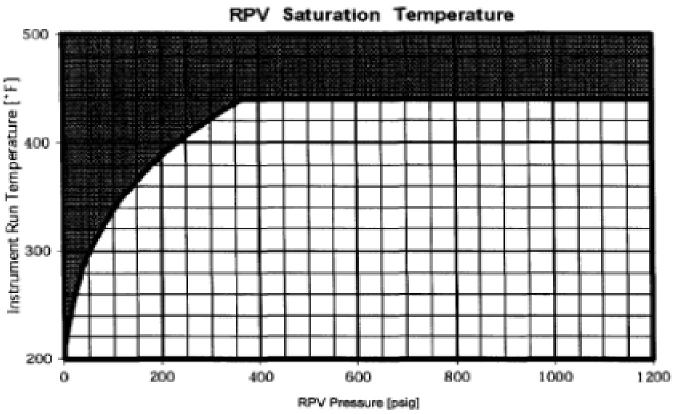
The following conditions are present:

- Drywell temperature: 405F
- RPV pressure: 225 psig
- Narrow Range level instruments: 165 inches and steady
- Wide Range level instruments: 32 inches and lowering
- Fuel Zone level indicator: Erratic, upscale and downscale
- Fuel Zone level recorder: De-energized
- Refuel Zone instrument: 300 inches and rising slowly
- EPIC: "locked-up", not updating

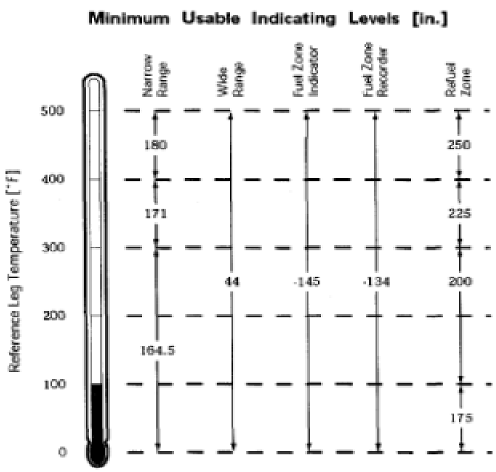
Which of the one of the following choices correctly lists which RPV level instrument(s) (if any) is\are indicating a valid reading?

Valid RPV level instrument(s)

A.	Narrow Range
B.	Refuel Zone
C.	Wide Range
D.	None



RPV Saturation Temperature
Instrument Run Temperature [°F]
500
400
300
200
0
200
400
600
800
1000
1200
RPV Pressure [psig]



Minimum Usable Indicating Levels [in.]
Narrow Range
Wide Range
Fuel Zone Indicator
Fuel Zone Recorder
Refuel Zone
Reference Leg Temperature [°F]
500
400
300
200
100
0
180
171
164.5
44
-145
-134
250
225
200
175

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EK3.02
	Importance Rating	3.5

Low Suppression Pool Wtr Lvl

Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: HPCI operation: Plant-Specific

Proposed Question: #15

The Plant has experienced a Seismic event. Torus water level cannot be maintained above 10.75 feet.

Which one of the following states the reason why EOP-4 (Primary Containment Control) directs the High Pressure Coolant Injection (HPCI) system tripped?

HPCI is tripped to prevent…

A. over-pressurizing the Torus.

B. localized boiling in the Torus.

C. vortex damage to the HPCI pump.

D. exceeding the Heat Capacity Temperature Limit (HCTL).

Proposed Answer: A

Explanation: HPCI steam discharge pipe becomes uncovered < 10.75 feet.

<u>EOP-4 Bases states</u>: (EO-4.05) Operation of the HPCI System with its exhaust discharge device not submerged will directly pressurize the torus. HPCI operation is therefore secured as required to preclude the occurrence of this condition. The consequences of not doing so may extend to failure of the primary containment from over pressurization, and thus HPCI must be secured irrespective of adequate core cooling concerns.

<u>HPCI Lesson Plan</u>:

Primary Containment:

a. HPCI Turbine exhausts to the Torus, below the normal water level.
b. Low level in Torus will result in exhaust steam filling the free airspace of the Torus & raising Torus pressure.
c. EOP-4 provides direction for control of HPCI on low Torus level.

<u>OP-15</u>:

Operating HPCI with suction from the torus and with torus water level below 10.4 feet could cause vortexing.

Technical Reference(s): EOP-4, OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11E 4.05, SDLP-23 1.10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments: DK 4/23: question format slightly changed due to NRC review \ suggestion.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK2.03
	Importance Rating	4.2

Reactor Low Water Level

Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Low pressure core spray

Proposed Question: #16

At time 0800, the Plant experienced a small loss of coolant accident in the Drywell.

Per EOP-2, (RPV Control), Residual Heat Removal (RHR) and Core Spray (CS) were Terminated and Prevented per EP-5, Termination and Prevention of RPV Injection.

At time 0900, the leak worsened and the following conditions are present:

- RPV level: 57 inches and lowering
- RPV pressure: 600 psig and lowering

Which one of the following states the response of the Core Spray system at time 0900?

	Core Spray Pumps	Core Spray Injection Valves
A.	Running	Open
B.	NOT running	Closed
C.	Running	Closed
D.	NOT running	Open

Proposed Answer: B

Explanation: Core Spray auto initiation signals are ≥2.7 psig or ≤59.5 inches. At time 0800 Core Spray was T&P'd. That means only the 2.7 psig signal was present and RPV level was not an issue at that time.

EOP-2 states: IF a high drywell press ECCS initiation signal exits THEN prevent injection from those CS and RHR pumps not required to assure adequate core cooling before depressurizing below their maximum injection pressures (EP-5).

EP-5 T&P of CS consists of:
Place 14MOV-11A AUTO ACTUATION BYPASS SW 14A-S16A switch in BYPASS.
Verify white 14MOV-11A AUTO ACTUATION BYPASS LT 14A-DS35A light is on.
Ensure closed OUTBD INJ VLV 14MOV-11A.
Ensure PMP 14P-1A is stopped.

Core Spray pumps will not auto restart after T&P is performed (manual start only).

Core Spray Injection Valves, 14MOV-11A (B) and 12A (B).
Outboard valves, 11A (B) are normally open and inboard valves, 12A (B) are normally closed.
Both valves in each system receive an automatic open signal upon receipt of a CSP initiation signal in conjunction with a Reactor pressure low signal (≤450 psig).

Technical Reference(s): EOP-2, EP-5, OP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14 1.05, 1.15

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: Changed to B correct based on validation feedback and research of electrical prints proves B to be correct.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK3.05
	Importance Rating	3.2

SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Cold shutdown boron weight

Proposed Question: #17

The Plant was operating at power when an automatic scram signal was received.

The following conditions **were** initially present:

- Control Rods: Fifty (50) at Position 48
- Rx power: 14% on APRMs
- Rx pressure: 920 psig controlled with EHC
- Rx level: 182 inches and lowering
- SLC Tank level: 82%

The following conditions are **now** present:

- Control Rods: Thirty (30) at Position 48
- Rx power: IRMs on Range 4 and lowering
- Rx pressure: 900 psig controlled with EHC
- Rx level: 95 inches and steady
- SLC Tank level: 34% and lowering

Based on the above conditions,

(1) is a cooldown of 85°F\hr now allowed per EOP-3 (Failure to Scram)
and
(2) why or why not?

	(1)	(2)
A.	Yes	RPV level is less than 100 inches
B.	No	IRMs are **NOT** less than Range 2
C.	Yes	Cold shutdown Boron weight has been injected
D.	No	All Control Rods are **NOT** inserted to or beyond Position 02

Proposed Answer: C

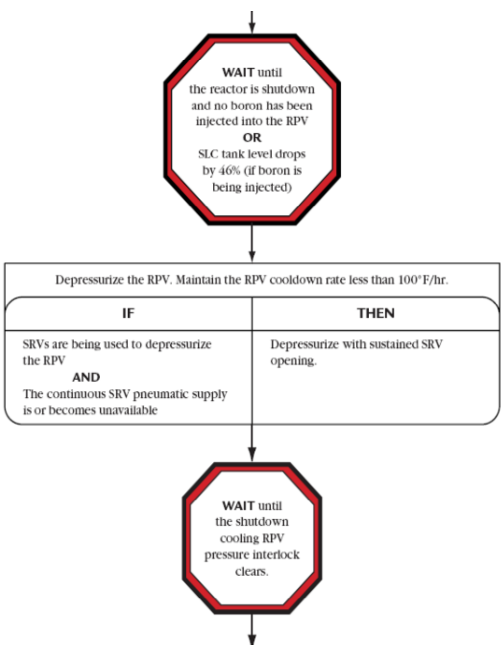
Explanation: RPV depressurization and cool down may not proceed until at least one of four conditions is satisfied:

- All control rods are inserted to or beyond position Maximum Subcritical Banked Withdrawal Position
- It has been determined that the reactor will remain shutdown under all conditions without boron
- Cold Shutdown Boron Weight of boron have been injected into the RPV
- The reactor is shutdown and no boron has been injected into the RPV.

The CSBW is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions. This weight is utilized to assure the reactor will remain shutdown irrespective of control rod position or RPV temperature.

If any amount of boron less than the CSBW has been injected into the RPV, cool down is not permitted unless it can be determined that control rod insertion alone assures the reactor will remain shutdown under all conditions. The core reactivity response from cool down in a partially borated core is unpredictable and subsequent EOP steps may not prescribe the correct actions for such conditions if criticality were to occur.

46% of the SLC tank must be injected before CSBW is reached; then a cooldown <100F\hr can commence.



Technical Reference(s): EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11D 1.07

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EA1.01
	Importance Rating	3.9

High Off-site Release Rate

Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Stack Gas Monitoring System: Plant-Specific

Proposed Question: #18

The Plant was operating at full power when Annunciators 09-3-2-38 (OFF GAS RAD MON HI-HI) and 09-3-2-10 (OFF GAS TIMER INITIATED) alarm.

Seventeen minutes later, Off-gas flow rate is reported to be 120 scfm and steady.

Which one of the following states the Operator action that is required per AOP-3 (HIGH ACTIVITY IN REACTOR COOLANT OR OFF-GAS)?

A. Open 28AOV-114A&B (SJAE AIR PURGE) to provide dilution air

B. Trip 38PMP-2A&B (CNDSR AIR RMVL PMP) to lower Off-gas flow rate

C. Begin lowering Reactor power per RAP-7.3.16 in an attempt to clear Annunciators

D. Close 01-107AOV-100 (OFF GAS DISCH TO STACK) to isolate the Off-gas system

Proposed Answer: D

Explanation: Annunciators 09-3-2-38 (OFF GAS RAD MON HI-HI) and 09-3-2-10 (OFF GAS TIMER INITIATED) begin a 15 minutes timer. After 15 minutes, if the HI-Hi condition is not clear, 01-107AOV-100 (OFF GAS DISCH TO STACK) should close to isolate the Off-gas system. With the conditions given in the stem, Off-gas flow rate of 120 scfm after 20 minutes means that 107AOV-100 has failed to close. AOP-3 provides direction to ensure closed 107AOV-100 and if it fails to close, then close other valves in the system (113A&B).

Technical Reference(s): AOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-38 1.14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AK3.04
	Importance Rating	2.8

Plant Fire On Site

Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site

Proposed Question: #19

A fire in the Relay Room has resulted in a Control Room evacuation.

AOP-28 (OPERATION DURING PLANT FIRES) and AOP-43 (PLANT SHUTDOWN FROM OUTSIDE THE CONTROL ROOM) are being executed.

AOP-43 directs the following valves to be closed within two hours:

- 27CAD-905 (DRYWELL PCV AND INSTR NITROGEN BACKUP ISOL VALVE)
- 27AOV-128A&B (CAD TRAIN A&B NITROGEN MAKE-UP SUPPLY VALVE)

Which one of the following states the reason why the above valves are to be shut?

A. The Nitrogen supply to SRVs may be depleted

B. The Nitrogen supply to MSIVs may be depleted

C. Unintentional Nitrogen pressurization of the Torus may occur

D. Unintentional Nitrogen pressurization of the Drywell may occur

Proposed Answer: A

Explanation: A Caution in AOP-43 states:

> **CAUTION**
>
> Failure to perform Steps F.3 through F.6 within 2 hours could result in the following:
>
> - East Crescent area temperature exceeding 110°F
> - Depletion of instrument N2 to the SRVs
> - Battery Room B/Battery Charger Room B temperature exceeding 110°F

F.4 Perform the following in CAD Building to preclude loss of instrument nitrogen to SRVs:

F.4.1 Close 27CAD-905 (drywell PCV and instr nitrogen backup isol valve).

NOTE: 27AOV-128A is inside 27NS-CA CAD A NITROGEN SUPPLY INSTRUMENT CABINET.

F.4.2 Fail close 27AOV-128A (CAD train A nitrogen make-up supply valve) as follows:

a. Close 27CAD-54A (27AOV-128A nitrogen supply isol valve).

b. Open petcock on bottom of 27PCV-128A (27AOV-128A nitrogen supply press regulator).

c. Verify closed 27AOV-128A.

NOTE: 27AOV-128B is inside 27NS-CB CAD B NITROGEN SUPPLY INSTRUMENT CABINET.

F.4.3 Fail close 27AOV-128B (CAD train B nitrogen make-up supply valve) as follows:

a. Close 27CAD-54B (27AOV-128B nitrogen supply isol valve).

b. Open petcock on bottom of 27PCV-128B (27AOV-128B nitrogen supply press regulator).

c. Verify closed 27AOV-128B.

Technical Reference(s): AOP-28, AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP SDLP-29 1.10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 2.4.11
	Importance Rating	4.0

Generator Voltage and Electric Grid Disturbances

Knowledge of abnormal condition procedures.

Proposed Question: #20

The Plant has just commenced a Tech Spec required shutdown. Current Rx power is 75%. Mode 2 is scheduled to be entered in 6 hours.

The following conditions occur:

- Strong thunderstorm activity is in the area
- Power Control reports that a complete loss of both 115KV lines is imminent

Per AOP-72 (115KV GRID LOSS, INSTABILITY OR DEGRADATION), when should HPCI and\or RCIC and the EDGs be started?

A. As soon as possible

B. Just prior to the Scram

C. Just after the Scram

D. When (if) they are required following the Scram

Proposed Answer: B

Explanation: Conditions given place the Plant in a condition of shutting down the Unit without offsite power. AOP-72 directs Rx power lowered to 50-70 MWe then inserting a manual scram and shutting the MSIVs. Per AOP-72, Section G Note: establishing HPCI, RCIC and EDG operation just prior to initiating reactor scram will minimize heat input to containment and minimize low load operation of the EDGs.

Technical Reference(s): AOP-72

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: Enhanced stem to clarify where in shutdown process the Plant is based on validation comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295002 AK1.03
	Importance Rating	3.6

Loss of Main Condenser Vacuum

Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM: Loss of heat sink

Proposed Question: #21

The Plant was operating at full power when it experienced a loss of all Circulating Water Pumps.

The following conditions exist:

- Rx power: 62% lowering slowly
- RPV pressure: 1015 psig steady
- RPV level: 120 inches lowering slowly
- Main Turbine: On-line

Which one of the following lists the appropriate mitigating strategy for the above conditions?

A. Shut the MSIVs due to the loss of the Ultimate Heat Sink

B. Maintain the Main Turbine on-line until it trips on low vacuum

C. Trip the Main Turbine to reduce Main Condenser vacuum loss

D. Install MSIVs jumpers to continue steam flow to the Main Condenser

Proposed Answer: A

Explanation: Conditions given are a situation of a Failure to scram compounded with a loss of Circulating Water.
Use of the Main Condenser is preferred over SRV operation which heats up the Torus however: Per EP-1: During EOP-3, closure of MSIVs is only permissible if one or more of the following conditions exist:

- Main steam line break
- All circulating water pumps have tripped
- Loss of main condenser vacuum
- IF EOP-3 override requiring use of main condenser is not in effect, ***AND*** either of the following conditions exist: EOP-6 directs isolation OR RPV cooldown continues after minimizing steam loads.

Technical Reference(s): EP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11D 1.07

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295009 AA2.01
	Importance Rating	4.2

Low Reactor Water Level

Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor water level

Proposed Question: #22

The Plant has experienced a loss of coolant accident.

Core Spray Pump (CSP) 'A' is the only injecting source of water.

- RPV level: -10 inches and lowering
- CSP 'A' flow rate: 5075 gpm
- Safety Relief Valves: Seven open

Based on the above conditions, which one of the following describes the status of "Adequate Core Cooling" (ACC)?

ACC…

A. is NOT presently assured.

B. is presently assured. ACC will be first lost if RPV level reaches -19 inches.

C. is presently assured. ACC will be first lost if RPV level reaches -31.5 inches.

D. is presently assured. ACC will be first lost if RPV level reaches -44.5 inches.

Proposed Answer: D

Explanation: EP-1 defines ACC:
Adequate Core Cooling
A. Heat removal from the reactor sufficient to prevent rupturing the fuel clad. Submergence is the preferred mechanism for cooling the core. The core is adequately cooled by submergence when it can be determined that RPV water level is at or above the top of active fuel.
Adequate spray cooling is provided, assuming a bounding axial power shape, when design spray flow requirements for the Core Spray System are satisfied and RPV water level is at or above the elevation of the jet pump suctions. **(4725 gpm and -44.5 inches)**
Steam cooling is relied upon only if RPV water level cannot be restored and maintained above TAF, cannot be determined, or must be intentionally lowered below TAF. The covered portion of the core remains cooled by boiling heat transfer which generates the steam that cools the uncovered portion.
Steam cooling will maintain the hottest peak clad temperature below: Steam Cooling with injection - < 1500F
Steam Cooling without injection - < 1800F
The "adequate core cooling" state can be defined only within the context of the Emergency Operating Procedures. Once conditions requiring entry of the Severe Accident Operating Guidelines (SAOGs) exist, a normal core configuration can no longer be assumed and the same criteria cannot be applied. While one of
the objectives of the SAOGs is to submerge the core or core debris, restoring RPV water level to above the top of the active fuel in accordance with the RPV and Primary Containment Flooding Strategies do not necessarily
re-establish adequate core cooling.

Technical Reference(s): EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT 301.11C 2.04

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295015 AK3.01
	Importance Rating	3.4

Incomplete SCRAM

Knowledge of the reasons for the following responses as they apply to INCOMPLETE SCRAM: Bypassing rod insertion blocks

Proposed Question: #23

The Plant was returning to full power following a Rod Sequence Exchange, when a Reactor scram occurred.

The following conditions exist:

- Mode Switch: SHUTDOWN
- Rx power: 10% on all APRMs
  Upscale on IRM range 9
- Control Rods: Scattered between Full In and Position 48

EP-3 (BACKUP CONTROL ROD INSERTION) is in progress and the Rx Scram has been reset.

Based on these conditions, which one of the following describes the need to bypass the Rod Worth Minimizer (RWM)?

The RWM…

A. MUST be bypassed because IRM upscale Rod Blocks are being enforced

B. does NOT have to be bypassed because Rx power is in the Transition Zone

C. MUST be bypassed because the RWM is invoking an Insert Error Control Rod Block

D. does NOT have to be bypassed because the Emergency Rod In switch will insert Control Rods

Proposed Answer: C

Explanation: The RWM ensures the reactor operator adheres to a predetermined sequence of control rod withdrawals or insertions when the reactor is operating at low power levels. Given the Rod pattern is scattered, Insert Blocks will be active and prevent the Operator from inserting Control Rods during the ATWS. Therefore, it must be bypassed to allow rod insertion. Transition zone is between 16% and 35% core thermal power. While in the transition zone, all normal RWM rod block actions are removed. Displays are still active, with exception of withdraw error windows which will not be displayed in the transition zones. Low Power Setpoint: If steam flow goes < 16%, the RWM is returned to operation.

Technical Reference(s): OP-64

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03D 1.05, 1.08, 1.15

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295020 AK2.01
	Importance Rating	3.6

Inadvertent Cont. Isolation

Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following:

Main steam system

Proposed Question: #24

The Plant was operating at full power when I&C inadvertently caused a PCIS Group 1 Isolation to be initiated on the **Div II side ONLY**.

Which one of the following choices lists the expected response of **(1)** the inboard Main Steam Isolation Valves [MSIVs] and **(2)** the outboard MSIVs?

	(1) inboard MSIVs	**(2) outboard MSIVs**
A.	Open	Open
B.	Open	Close
C.	Close	Open
D.	Close	Close

Proposed Answer: A

Explanation: Group I Isolations: Logic: Consists of four sub-channels in a 1 out of 2 taken twice arrangement for MSIVs. De-energize to actuate. Group I Isolation logic signals: Reactor Vessel Water Level Lo-Lo-Lo K1A, B, C, D (≥18 inches): MSIVs Inboard & Outboard **(1 out of 2 taken twice)**.

Group II isolation is a 2 out of 2 taken once logic arrangement.

Technical Reference(s): OP-1, AOP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16C 1.06

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295022 AA1.03
	Importance Rating	2.7

Loss of CRD Pumps

Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: Recirculation system: Plant-Specific

Proposed Question: #25

The Plant is operating at full power with CRD pump ‘B’ tagged out of service for maintenance.

The following occurs:

- CRD pump ‘A’ trips

Which one of the following Reactor Recirculation Pump components \ parameters will require increased monitoring?

A. Pump seals

B. Motor current

C. Pump vibration

D. Motor temperatures

Proposed Answer: A

Explanation: The CRD system provides clean, cool water (called mini-purge) to the RWR pump seals. This mini-purge flushes or maintains the seals clean and cool. A loss of this mini-purge will cause the seals to over-heat and degrade.

Technical Reference(s): OP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.09

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295029 EK2.06
	Importance Rating	3.4

High Suppression Pool Wtr Lvl

Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: SRVs and discharge piping

Proposed Question: #26

The Plant has experienced a transient with following:

- Multiple control rods failed to insert following a Reactor scram.
- Reactor pressure is 1200 psig.
- Torus water level is 17.5'.

Which one of the following describes the status of the SRV Tail Pipe Level Limit and the associated basis for the SRV Tail Pipe Level Limit at this Reactor pressure?

The SRV Tail Pipe Level Limit has ___(1)___. The basis for the SRV Tail Pipe Level Limit at this Reactor pressure is to prevent damage to the ___(2)___.

	___(1)___	___(2)___
A.	been exceeded	Containment in the event of a LOCA
B.	been exceeded	SRV tail pipes upon SRV opening
C.	NOT been exceeded	Containment in the event of a LOCA
D.	NOT been exceeded	SRV tail pipes upon SRV opening

Proposed Answer: B

Explanation: The SRV Tail Pipe Level Limit (SRVTPLL) is limited to 17.85 feet between 0 and 1165 psig. The limiting component in this range of Reactor pressures is the containment due to the reduced air space to cushion the pressure surge should a LOCA occur. The SRVTPLL lowers at Reactor pressures above 1165 psig. The combination of 1200 psig and Torus water level of 17.5' exceeds the limit. The limiting component in this range of Reactor pressures is the SRV tailpipe due to high clearing loads upon SRV actuation.

A. Incorrect – The Containment is the limiting component of the SRVTPLL at Reactor pressures below 1165 psig.
C. Incorrect – The SRVTPLL has been exceeded. The Containment is the limiting component of the SRVTPLL at Reactor pressures below 1165 psig.
D. Incorrect – The SRVTPLL has been exceeded.

Technical Reference(s): EOP-11, MIT-301.11B

Proposed references to be provided to applicants during examination: SRV Tail Pipe Level Limit Curve

Learning Objective: MIT-301.11B 1.01

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	500000 EA1.02
	Importance Rating	3.3

High CTMT Hydrogen Conc.

Ability to operate and / or monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: Primary containment oxygen instrumentation

Proposed Question: #27

The Plant has experienced a Loss of Coolant Accident with the following:

- H2/O2 monitors have been reinitialized.
- H2/O2 indications on Relay Room panels 27PCX-101A (B) are not displaying properly.

Which one of the following identifies alternate location(s) for H2/O2 indications?

A. EPIC, only

B. Remote Shutdown Panel, only

C. H2/O2 Analyzer Cabinets, 27PCA-101A(B), on RB Elevation 300’, only

D. EPIC and H2/O2 Analyzer Cabinets, 27PCA-101A(B), on RB Elevation 300’

Proposed Answer: D

Explanation:

A. Incorrect – EPIC and 27PCA-101A(B) include H2/O2 indication.
B. Incorrect – The Remote Shutdown Panel does not have H2/O2 indication.
C. Incorrect – EPIC display CAS 1 includes H2/O2 indication.

Technical Reference(s): SDLP-16B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16B 1.11.a.6, 1.11.a.7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 5/22/12 – Edited wording of question and choices to eliminate T/F question, based on NRC comment.

TRH 6/1/12 - Edited choice from “None” to “Remote Shutdown Panel, only”, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 A4.09
	Importance Rating	4.1

RHR/LPCI: Injection Mode

Ability to manually operate and/or monitor in the control room: System flow

Proposed Question: #28

A Loss of Coolant Accident results in the following:

Time (minutes)	Event
0	The Reactor scrams on high Drywell pressure.
1	Reactor water level is 126.5” and lowering. Reactor pressure is 900 psig and lowering.
2	Reactor water level is 105.4” and lowering. Reactor pressure is 700 psig and lowering.
3	Reactor water level is 59.5” and lowering. Reactor pressure is 550 psig and lowering.
4	Reactor water level is 0” and lowering. Reactor pressure is 450 psig and lowering.
5	Reactor water level is 10” and rising. Reactor pressure is 375 psig and lowering.

Drywell pressure remains above 2.7 psig and Terminate and Prevent has **NOT** been performed.

Which one of the following identifies when 10MOV-27A(B), LPCI OUTBD INJ VLV, and 10MOV-66A(B), HX A(B) BYP VLV, can first be throttled to control LPCI system flow?

	10MOV-27A(B) Can First Be Throttled at Time…	10MOV-66A(B) Can First Be Throttled at Time…
A.	8 minutes	3 minutes
B.	8 minutes	6 minutes
C.	9 minutes	3 minutes
D.	9 minutes	6 minutes

Proposed Answer: C

Explanation: LPCI receives an auto-initiation signal on either high Drywell pressure (2.7 psig) or low-low-low Reactor water level (59.5”). LPCI injection valves receive an open signal when one of the auto-initiation signals is present plus Reactor pressure lowers to 450 psig. 10MOV-66A(B) receive an open signal on system initiation. This open signal seals-in for 3 minutes. Therefore 10MOV-66A(B) received an open signal at time 0 minutes, and will become throttleable at time 3 minutes. 10MOV-27A(B) received an open signal at time 4 minutes, and this open signal seals-in for 5 minutes. Therefore 10MOV-27A(B) will become throttleable at time 9 minutes.

A. Incorrect – 10MOV-27A(B) have a sealed-in open signal until time 9 minutes.
B. Incorrect – 10MOV-27A(B) have a sealed-in open signal until time 9 minutes. 10MOV-66A(B) can be first throttled at time 3 minutes.
D. Incorrect – 10MOV-66A(B) can be first throttled at time 3 minutes.

Technical Reference(s): OP-13A, ARP-09-3-1-27, DBD-10

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.05.a.1.c, 1.05.a.1.d

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: 6/28/12 - Added “terminate and prevent has **NOT** been performed” based on validator comments

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 2.1.32
	Importance Rating	3.8

Shutdown Cooling

Ability to explain and apply system limits and precautions.

Proposed Question: #29

The Plant is shutdown for a refueling outage with the following:

- RHR pump B is operating in the Shutdown Cooling (SDC) lineup.
- All other RHR pumps are secured.
- RHR pump B flow is 4000 gpm.

Which one of the following describes how RHR pump B flow should be adjusted?

Flow should be…

A. lowered to prevent pump runout.

B. lowered to prevent valve flow erosion.

C. raised to prevent high pump vibration.

D. raised to prevent cycling of the minimum flow valve.

Proposed Answer: C

Explanation: OP-13D, RHR – Shutdown Cooling, Precaution C.2.4 states:

“Operation of any single RHR pump at flows less than 6500 gpm or operation of any two RHR pumps at flows less than 13000 gpm should be minimized to prevent high pump vibration.”

A. Incorrect – 4000 gpm is below the minimum of 6500 gpm, therefore flow should be raised.
B. Incorrect – 4000 gpm is below the minimum of 6500 gpm, therefore flow should be raised.
D. Incorrect – The minimum flow valve cycles in the range of 1250-1450 gpm. At 4000 gpm, the minimum flow valve would be closed.

Technical Reference(s): OP-13D

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.13.d

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 K3.01
	Importance Rating	4.0

HPCI

Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE COOLANT INJECTION SYSTEM will have on following: Reactor water level control: BWR-2,3,4

Proposed Question: #30

A Loss of Coolant Accident has resulted in the following:

- The Reactor has scrammed.
- MSIVs are closed.
- Condensate, Feedwater, and RCIC are unavailable.
- HPCI is injecting 3500 gpm in AUTO.
- Reactor water level is 200” and stable.
- Reactor pressure is 850 psig and slowly lowering.

Then, the flow input signal to the HPCI pump controller fails upscale.

Which one of the following describes the effect of this malfunction on Reactor water level and the availability of HPCI?

	Reactor water level…	HPCI…
A.	lowers.	is no longer available for injection.
B.	lowers.	may be used for injection in MANUAL.
C.	rises.	is no longer available for injection.
D.	rises.	may be used for injection in MANUAL.

Proposed Answer: B

Explanation: The HPCI controller varies turbine speed to maintain sensed flow at the demanded setpoint when in AUTO. If the sensed flow goes up, the HPCI controller lowers turbine speed in an attempt to lower sensed flow. Since the flow signal is failed upscale, the HPCI controller will run back to minimum demand signal with no effect on sensed flow. Actual flow will lower, causing Reactor water level to lower. The HPCI controller does not use the sensed flow signal when in the MANUAL mode. Therefore HPCI is still available in MANUAL.

A. Incorrect – HPCI is still available for injection with the flow controller in MANUAL.
C. Incorrect – Reactor water level lowers. HPCI is still available for injection with the flow controller in MANUAL.
D. Incorrect – Reactor water level lowers.

Technical Reference(s): OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.09.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: Based on validator comment, enhanced description of flow signal failure. 7/3/DK underlined flow input signal.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 A3.03
	Importance Rating	3.9

HPCI

Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including: System lineup: BWR-2,3,4

Proposed Question: #31

A Plant transient results in the following:

Time (min)	Event
0	HPCI initiates on low Reactor water level.
5	Reactor water level rises to the HPCI high level trip setpoint.
10	The HPCI high level trip signal clears.

Which one of the following describes the response of HPCI at time 5 minutes and 10 minutes?

	Response at 5 Minutes	Response at 10 Minutes
A.	23HOV-1, HPCI Turbine Stop Valve, closes.	23HOV-1, HPCI Turbine Stop Valve, remains closed.
B.	23HOV-1, HPCI Turbine Stop Valve, closes.	23HOV-1, HPCI Turbine Stop Valve, re-opens.
C.	23MOV-14, HPCI Turbine Steam Supply Isolation Valve, closes.	23MOV-14, HPCI Turbine Steam Supply Isolation Valve, remains closed.
D.	23MOV-14, HPCI Turbine Steam Supply Isolation Valve, closes.	23MOV-14, HPCI Turbine Steam Supply Isolation Valve, re-opens.

Proposed Answer: A

Explanation: HPCI receives a trip signal on high Reactor water level (222.5”). The trip signal causes 23HOV-1 to close, but does not re-position 23MOV-14. When the high level condition clears, 23HOV-1 remains closed until either (1) the trip signal is manually reset and an initiation signal (manual or automatic) is present or (2) Reactor water level lowers to the low-low level initiation setpoint (126.5”).

B. Incorrect – 23HOV-1 remains closed when the high level signal clears.
C. Incorrect – 23MOV-14 remains open.
D. Incorrect – 23MOV-14 remains open.

Technical Reference(s): ARP-09-3-3-25, OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.05.a.11, 1.05.a.18

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K1.02
	Importance Rating	3.4

LPCS

Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Torus/suppression pool

Proposed Question: #32

A Loss of Coolant Accident has occurred with the following:

- The Core Spray Loop A suction source is to be swapped from the Torus to the Condensate Storage Tanks (CSTs) due to indications of strainer blockage.
- 14MOV-7A, CSP Pump A Suct From Suppr Pool Isol Valve, is currently open.
- 14CSP-8A, Core Spray Pump A Suct From CST A & B Isol Valve, is currently closed.

Which one of the following describes the sequence of operation for the Core Spray Loop A suction source swap?

A. Open 14CSP-8A and then close 14MOV-7A, to ensure continuity of Core Spray Loop A suction.

B. Open 14CSP-8A and then close 14MOV-7A, to ensure proper differential pressure for opening 14CSP-8A.

C. Close 14MOV-7A and then open 14CSP-8A, to ensure the CSTs are not drained to the Torus.

D. Close 14MOV-7A and then open 14CSP-8A, to ensure the Torus is not drained to the CSTs.

Proposed Answer: C

Explanation: OP-14 Precaution C.2.5 states:

"CSP Pump Suction From Suppression Pool Isolation Valve 14MOV-7A or 14MOV-7B and the associated Core Spray Pump Suction From CST A & B Isolation Valve 14CSP-8A or 14CSP-8B shall not be open at the same time to prevent draining CSTs to the Torus."

OP-14 section G.9 provides the procedure for swapping Core Spray Loop A suction from the Torus to the CSTs. This section implements Precaution C.2.5 by closing 14MOV-7A and then opening 14CSP-8A.

A. Incorrect – 14MOV-7A is closed before 14CSP-8A is opened.
B. Incorrect – 14MOV-7A is closed before 14CSP-8A is opened.
D. Incorrect – This would result in draining the CSTs to the Torus.

Technical Reference(s): OP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14 1.13.c

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 K5.07
	Importance Rating	2.7

SLC

Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: Tank heater operation

Proposed Question: #33

The Plant is operating at 100% power with the following:

- Annunciator 09-3-3-10, SLC TK OR SUCT LINE TEMP HI OR LO, alarms.
- An Operator in the field reports Standby Liquid Control (SLC) tank temperature is 67^{o}F.

Which one of the following describes the implication of this SLC tank temperature?

The SLC tank heater should be…

A. OFF to ensure accurate tank level indication.

B. OFF to ensure adequate boron solution concentration.

C. ON to ensure accurate tank level indication.

D. ON to ensure adequate boron solution concentration.

Proposed Answer: D

Explanation: The SLC tank heater controller should maintain tank temperature 82-90^{o}F. At or below 82^{o}F, the tank heater should be on. 67^{o}F is the low tank temperature setpoint. The purpose of the tank heater is to maintain solution temperature high enough such that boron does not precipitate out of solution, which would lead to lowering solution concentration.

A. Incorrect – Since temperature is low, the heater should be on. The concern with low temperature is for boron solution concentration.
B. Incorrect – Since temperature is low, the heater should be on.
C. Incorrect – The concern with low temperature is for boron solution concentration.

Technical Reference(s): ARP-09-3-3-10, OP-17

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-11 1.05.a.6, 1.05.a.7, 1.05.b.3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 K6.03
	Importance Rating	3.2

SLC

Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY LIQUID CONTROL SYSTEM: A.C. power

Proposed Question: #34

An ATWS has occurred with the following:

- MCC 162 has de-energized due to a fault.
- Standby Liquid Control (SLC) injection is being initiated from Panel 09-3.

Which one of the following describes the impact of the electrical fault on SLC injection capability?

A. Full design flow is still available through one pump and two squib valves.

B. Full design flow is still available through one pump and one squib valve.

C. Only 50% of design flow is available through one pump and two squib valves.

D. Only 50% of design flow is available through one pump and one squib valve.

Proposed Answer: B

Explanation: MCC 162 provides power for the following SLC components:

- SLC pump B
- SLC B squib valve
- SLC storage tank heater

With the loss of MCC 162, only SLC pump A and SLC A squib valve are available for injection. The SLC pumps are redundant, 100% capacity positive-displacement pumps that provide a flow rate of 50 gpm. The SLC squib valves are redundant, parallel valves each capable of passing the 50 gpm flow rate provided by the positive-displacement SLC pumps.

A. Incorrect – Only one squib valve is available.
C. Incorrect – SLC can still provide full flow, and only one squib valve is available.
D. Incorrect – SLC can still provide full flow.

Technical Reference(s): UFSAR Section 3.9, OP-17

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 A1.08
	Importance Rating	3.4

RPS

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: Valve position

Proposed Question: #35

The Plant is operating at 85% power with the following:

- A manual scram is required due to high Turbine vibrations.
- MANUAL SCRAM A and MANUAL SCRAM B pushbuttons are depressed.
- The Mode Switch is placed in SHUTDOWN.
- Reactor water level reaches a low value of 140” before recovering to the normal band.
- AOP-1, Reactor Scram, is being executed.
- SDIV HI LVL TRIP keylock switch is placed in BYPASS.
- The scram has **NOT** yet been reset.

Which one of the following lists the position of the backup scram valves and the SDIV vent and drain valves?

	Backup Scram Valves	SDIV Vent and Drain Valves
A.	Venting the scram air header	Open
B.	Venting the scram air header	Closed
C.	Supplying instrument air to the scram air header	Open
D.	Supplying instrument air to the scram air header	Closed

Proposed Answer: B

Explanation: The backup scram valves re-position on any scram signal to isolate the air supply to the scram air header and vent off air in the scram air header. The backup scram valves remain in this position until the scram is reset. The SDIV vent and drain valves close on any scram signal. The SDIV HI LVL TRIP keylock switch is placed in BYPASS to allow the scram to be reset with high water level in the SDIV. However, placing this switch in BYPASS does not re-position the SDIV vents and drains. The SDIV vents and drains re-open once the scram is reset.

A. Incorrect – SDIV vent and drain valves are closed.
C. Incorrect – Backup scram valves are venting the scram air header and SDIV vent and drain valves are closed.
D. Incorrect – Backup scram valves are venting the scram air header.

Technical Reference(s): AOP-1, OP-25

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.05.c.5, 1.05.c.7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 2.4.9
	Importance Rating	3.8

RPS

Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: #36

The Plant is shutdown with the following:

- RHR Loop A is in the Shutdown Cooling lineup.
- RHR Pump A is running and RHR Pump C is secured.
- Reactor water temperature is 100^{o}F and stable.
- Then, RPS Bus A de-energizes due to a fault and **CANNOT** be re-energized.

Which one of the following describes the effect on the Shutdown Cooling lineup?

A. RHR Pump A trips and RHR Pump C is unavailable, but the Shutdown Cooling lineup remains un-isolated. RHR Loop B may be used to re-establish Shutdown Cooling.

B. RHR Pump A trips, but the Shutdown Cooling lineup remains un-isolated. RHR Pump C may be started to re-establish Shutdown Cooling.

C. The Shutdown Cooling lineup isolates. RHR valves may be manually re-opened to re-establish Shutdown Cooling.

D. The Shutdown Cooling lineup isolates and **CANNOT** be re-established.

Proposed Answer: C

Explanation: The loss of RPS Bus A causes RHR valves 10MOV-17, 18, and 25A to close, which isolates the Shutdown Cooling lineup. AOP-30 provides guidance to re-open these valves (either locally or by jumpering the isolation signal) to re-establish Shutdown Cooling.

A. Incorrect – 10MOV-17, 18, and 25A close, which isolates the Shutdown Cooling lineup.
B. Incorrect – 10MOV-17, 18, and 25A close, which isolates the Shutdown Cooling lineup.
D. Incorrect – AOP-30 provides guidance to re-open valves (either locally or by jumpering the isolation signal) to re-establish Shutdown Cooling.

Technical Reference(s): AOP-59, AOP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.10.b

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: Enhanced D by adding manually for clarification based on validator comment.

6/28 - Swapped C and D position, got rid of some wording in new D, based on high miss rate.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K4.01
	Importance Rating	3.7

IRM

Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Rod withdrawal blocks

Proposed Question: #37

A Plant startup is in progress with the following:

- The Mode Switch is in START & HOT STBY.
- All Intermediate Range Monitor (IRM) detectors are fully inserted and indicating mid-scale on range 1.
- Then, a shorted contact causes IRM G detector to withdraw.

Which one of the following describes the resulting status of control rod blocks?

A. No rod block is received.

B. A rod block is received as soon as IRM G leaves the fully inserted position.

C. A rod block is received when IRM G exits the core region.

D. A rod block is received when IRM G indication lowers to below 2.5/125 of scale.

Proposed Answer: B

Explanation: With the Mode Switch in START & HOT STBY, a rod block is received if an IRM leaves the fully inserted position. The IRM downscale rod block is bypassed when an IRM is on range 1.

A. Incorrect – With the Mode Switch in START & HOT STBY, a rod block is received if an IRM leaves the fully inserted position.
C. Incorrect – The rod block is received earlier, as soon as the detector leaves the fully inserted position.
D. Incorrect – The IRM downscale rod block is bypassed when an IRM is on range 1. Also, the rod block is received earlier than a downscale would be detected.

Technical Reference(s): ARP-09-5-2-02, OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.07.b

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: 7/3/DK TR says leave as is.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 A1.06
	Importance Rating	3.3

IRM

Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including: Lights and alarms

Proposed Question: #38

A Plant startup is in progress with the following:

- The Mode Switch is in START & HOT STBY.
- All Intermediate Range Monitor (IRM) detectors are indicating mid-scale on range 5.
- Then, the IRM A mode switch is moved from OPERATE to STANDBY on Panel 09-12.

Which one of the following describes the resulting status of Panel 09-5 IRM A indicating lights?

	IRM A UPSC TR OR INOP light	IRM A BYP light
A.	Lit	Lit
B.	Lit	**NOT** lit
C.	**NOT** lit	Lit
D.	**NOT** lit	**NOT** lit

Proposed Answer: B

Explanation: When IRM A mode switch is taken out of OPERATE, an INOP signal is generated. This illuminates the IRM A UPSC TR OR INOP light on Panel 09-5, but not the IRM A BYP light.

A. Incorrect – The IRM A BYP light is not lit.
C. Incorrect – The IRM A UPSC TR OR INOP light is lit and the IRM A BYP light is not lit.
D. Incorrect – The IRM A UPSC TR OR INOP light is lit.

Technical Reference(s): ARP-09-5-2-52, 1.66-180

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.05.a.6.a, 1.11.a.7.a, 1.11.a.7.d

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 A3.03
	Importance Rating	3.6

Source Range Monitor

Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: RPS status

Proposed Question: #39

A Plant startup is in progress with the following:

- Special low-power physics testing is being conducted due to a new fuel design.
- The Mode Switch is in START & HOT STBY.
- The Source Range Monitor (SRM) Shorting Links are removed.
- Then, an inadvertent cold-water addition results in the following SRM indications:
  - SRM A: 75,000 cps
  - SRM B: 110,000 cps
  - SRM C: 90,000 cps
  - SRM D: 100,000 cps

Which one of the following describes the status of the Reactor Protection System (RPS)?

A. No scram signals have been received due to SRMs.

B. A half-scram signal has been received on RPS A due to SRMs.

C. A half-scram signal has been received on RPS B due to SRMs.

D. A full scram signal has been received due to SRMs.

Proposed Answer: A

Explanation: The SRM scram signal is only applicable with the associated RPS shorting links removed. With the shorting links removed, any SRM above 5×10^5 cps will cause a full scram. Since no SRM is above 5×10^5 cps, no scram signal is present.

B. Incorrect – No SRM has exceeded 5×10^5 cps.
C. Incorrect – No SRM has exceeded 5×10^5 cps.
D. Incorrect – No SRM has exceeded 5×10^5 cps.

Technical Reference(s): OP-16, SDLP-07B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.05.b.3, 1.05.c.e

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: changed SRM D value to 100000 so not too close to actual plant setpoint based on validator comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 K2.02
	Importance Rating	2.6

APRM / LPRM

Knowledge of electrical power supplies to the following: APRM channels

Proposed Question: #40

The Plant is operating at 100% power when Reactor Protection System (RPS) Bus A de-energizes due to a fault.

Which one of the following identifies the Average Power Range Monitors (APRMs) that remain available for monitoring Reactor power?

APRMs…

A. A, B, and C

B. D, E, and F

C. A, C, and E

D. B, D, and F

Proposed Answer: D

Explanation: RPS Bus A powers APRMs A, C, and E. RPS Bus B powers APRMs B, D, and F. With loss of RPS Bus A, only the RPS Bus B powered APRMs (B, D, and F) are available for monitoring Reactor power.

A. Incorrect – APRMs A and C are not available.
B. Incorrect – APRM E is not available.
C. Incorrect – APRMs A, C, and E are not available.

Technical Reference(s): OP-16, AOP-59

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07C 1.04.b

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 A1.01
	Importance Rating	3.7

RCIC

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: RCIC flow

Proposed Question: #41

The Plant is operating at 100% power with the following:

- ST-24R, RCIC Turbine Slow Roll Test (Mode 1), is being performed following RCIC maintenance.
- RCIC FLOW CNTRL 13FIC-91 is in BAL with the normal standby setpoint.
- The TURB TEST 13A-S20 switch is in TURB TEST.
- The TEST PWR 13A-S21 switch is in ON.
- RCIC is running with a flow rate of 300 gpm.

Then, a Reactor scram occurs and Reactor water level lowers to 120”.

Which one of the following describes the RCIC flow rate and method of control?

RCIC flow rate ___(1)___. RCIC flow is now controlled by manipulating the ___(2)___.

	(1)	(2)
A.	remains unchanged	TEST SPEED ADJUST potentiometer
B.	remains unchanged	RCIC FLOW CNTRL setpoint wheel
C.	lowers	RCIC FLOW CNTRL setpoint wheel
D.	rises	RCIC FLOW CNTRL setpoint wheel

Proposed Answer: D

Explanation: The RCIC turbine test circuit is provided to control RCIC turbine speed for system testing. It uses a potentiometer to supply the turbine speed signal vice the normal flow controller. However, if a RCIC auto-initiation signal occurs while in the test mode, the test circuit is automatically isolated and the normal flow controller takes over. Reactor water level less than 126.5” is a RCIC auto-initiation signal. Since initial RCIC flow was 300 gpm and normal flow controller setpoint is 400 gpm, RCIC flow will rise. The test potentiometer is now isolated from the circuit, thus further changes to RCIC flow will be made by adjusting the flow controller setpoint.

A. Incorrect – RCIC flow control swaps to the flow controller and rises to 400 gpm.
B. Incorrect – RCIC flow rises to the normal setpoint of 400 gpm.
C. Incorrect – RCIC flow rises to the normal setpoint of 400 gpm.

Technical Reference(s): ST-24R, OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13 1.05.a.9, 1.05.a.16

Question Source: Modified Bank Cooper 2002 NRC #13

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K3.02
	Importance Rating	4.5

ADS

Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Ability to rapidly depressurize the reactor

Proposed Question: #42

The Plant was operating at 100% power when the following events occurred:

- An un-isolable steam leak in the Reactor Building has led to the need for an Emergency RPV Depressurization.
- Equipment malfunctions caused eight (8) SRVs to fail closed.
- MSIVs closed due to an invalid isolation signal that has now cleared.
- Main Condenser vacuum is 25 inches Hg and steady.

Which one of the following describes the effect of the SRV failures on the ability to rapidly depressurize the Reactor, in accordance with EOP-2, RPV Control?

The Minimum Number of SRVs Required for Emergency Depressurization ___(1)___ and Turbine Bypass Valves ___(2)___ be used to rapidly depressurize the Reactor.

	(1)	(2)
A.	is **NOT** available	may
B.	is **NOT** available	may **NOT**
C.	is available	may
D.	is available	may **NOT**

Proposed Answer: A

Explanation: The Plant has 11 SRVs and the Minimum Number of SRVs Required for Emergency Depressurization is 5. With 8 SRVs failed closed, only 3 are available, which is less than the required 5. With less than 5 SRVs open, EOP-2 directs use of Group 2 Pressure Control Systems to rapidly depressurize the Reactor. Group 2 Pressure Control Systems include Turbine Bypass Valves. Allowance is given to re-open the MSIVs if closed and to override isolation signals if present.

B. Incorrect – EOP-2 directs use of Group 2 Pressure Control Systems to rapidly depressurize the Reactor. Group 2 Pressure Control Systems include Turbine Bypass Valves. Allowance is given to re-open the MSIVs if closed and to override isolation signals if present.
C. Incorrect – The Minimum Number of SRVs Required for Emergency Depressurization is 5. With 8 SRVs unavailable, only 3 remain available.
D. Incorrect – The Minimum Number of SRVs Required for Emergency Depressurization is 5. With 8 SRVs unavailable, only 3 remain available. EOP-2 directs use of Group 2 Pressure Control Systems to rapidly depressurize the Reactor. Group 2 Pressure Control Systems include Turbine Bypass Valves. Allowance is given to re-open the MSIVs if closed and to override isolation signals if present.

Technical Reference(s): EOP-2, MIT-301.11B

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11C 4.04

Question Source: Bank NMP1 2010 NRC #8

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K1.14
	Importance Rating	2.8

PCIS/Nuclear Steam Supply Shutoff

Knowledge of the physical connections and/or cause-effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: Containment drainage system

Proposed Question: #43

Which one of the following identifies the Primary Containment Isolation System (PCIS) Group that contains 20MOV-82, Drywell Floor Drain Sump Inboard Isolation Valve, and the parameter(s) that cause(s) isolation of this group?

	PCIS Group	Parameter(s) That Cause(s) Isolation Of This Group
A.	I	High Drywell pressure, only
B.	II	High Drywell pressure, only
C.	I	Low RPV water level and/or high Drywell pressure
D.	II	Low RPV water level and/or high Drywell pressure

Proposed Answer: D

Explanation: 20MOV-82, Drywell Floor Drain Sump Inboard Isolation Valve, is part of PCIS group II. PCIS group II isolates on either high Drywell pressure or low RPV water level.

A. Incorrect - 20MOV-82 is part of PCIS group II. Low RPV water level also isolates PCIS group II.
B. Incorrect -
C. Incorrect - 20MOV-82 is part of PCIS group II. Low RPV water level also isolates PCIS group II.

Technical Reference(s): AOP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16C 1.05.a.3, 1.09.d

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: 7/3/12 - Edited 2nd column question due to high validation miss rate.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 A2.01
	Importance Rating	3.0

SRVs

Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open vacuum breakers

Proposed Question: #44

A plant transient results in the following:

- SRV K opens due to high Reactor pressure and then closes.
- An SRV K vacuum breaker opens and sticks in the open position.

Which one of the following describes the consequence of the stuck open SRV vacuum breaker?

If SRV K opens again…

A. the pressure trapped in the discharge pipe would raise the operating setpoint of the SRV, resulting in delayed opening time.

B. the water that was drawn up into the discharge pipe would cause a water hammer to occur, causing possible damage to the SRV, piping, and Torus.

C. the steam passing through the SRV will be released directly into the Torus airspace, bypassing the pressure suppression function of the primary containment.

D. the steam passing through the SRV will be released directly into the Drywell airspace, producing primary containment conditions similar to a small break LOCA.

Proposed Answer: D

Explanation: After SRV operation, the vacuum breakers open to equalize pressure between the Drywell and tailpipes. Without vacuum breaker operation, condensation of steam in the tailpipe draws water from the Torus up into the tailpipe. Upon subsequent re-opening of the SRV, high forces would be experienced due to the clearing of the extra water from the tailpipe. With a stuck open vacuum breaker, subsequent SRV opening would admit steam directly to the Drywell airspace, resulting in rising Drywell temperature and pressure.

A. Incorrect – Pressure in the tailpipe will be equalized with Drywell pressure due to the stuck open vacuum breaker.
B. Incorrect – This is the negative effect of vacuum breakers failing closed.
C. Incorrect – The SRV tailpipe vacuum breakers connect to the Drywell airspace, not Torus.

Technical Reference(s): SDLP-02J

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.09.f

Question Source: Bank Cooper 2002 NRC #8

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 A4.01
	Importance Rating	3.8

Reactor Water Level Control

Ability to manually operate and/or monitor in the control room: All individual component controllers in the manual mode

Proposed Question: #45

A Plant startup is in progress with the following:

- Reactor Feed Pump Turbine (RFPT) A is in service.
- RFPT B is being started.

Which one of the following identifies the control to be manipulated to raise RFPT B speed when at 700 rpm and when at 2000 rpm, in accordance with OP-2A, Feedwater System?

When RFPT speed is at 700 rpm, manipulate RFPT B ___(1)___ to raise speed. When RFPT is at 2000 rpm, manipulate RFPT B ___(2)___ to raise speed.

	(1)	(2)
A.	FLOW CNTRL 06-84B	FLOW CNTRL 06-84B
B.	FLOW CNTRL 06-84B	MTR SPEED CHANGER
C.	MTR SPEED CHANGER	FLOW CNTRL 06-84B
D.	MTR SPEED CHANGER	MTR SPEED CHANGER

Proposed Answer: C

Explanation: At speeds less than 800 rpm, only the Motor Speed Changer (MSC) is capable of controlling. By 1800 rpm, the Motor Gear Unit (MGU) is in control. The MGU is controlled using RFPT B FLOW CNTRL 06-84B.

A. Incorrect - RFPT B FLOW CNTRL 06-84B will not work below 800 rpm.
B. Incorrect - RFPT B FLOW CNTRL 06-84B will not work below 800 rpm. Above 1800 rpm, RFPT B FLOW CNTRL 06-84B is used to control speed, with the Motor Speed Changer raised to the high stop.
D. Incorrect - Above 1800 rpm, RFPT B FLOW CNTRL 06-84B is used to control speed, with the Motor Speed Changer raised to the high stop.

Technical Reference(s): OP-2A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-06 1.05.a.7, 1.05.a.8

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: 7/3/DK added MTR to B2 (typo)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 K4.02
	Importance Rating	2.6

SGTS

Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Charcoal bed decay heat removal

Proposed Question: #46

The Plant has experienced a steam leak in the Reactor Building with the following:

Time (minutes)	Event
0	Both trains of Standby Gas Treatment (SBGT) automatically start.
60	SBGT train A is being secured. 01-125MOV-14A, TRAIN A INLET, is manually closed at the 09-75 Panel.

Which one of the following describes the position of 01-125MOV-100A, TRAIN A CLG VLV, at time 55 minutes and at time 65 minutes?

	Position of 01-125MOV-100A at Time 55 Minutes	Position of 01-125MOV-100A at Time 65 Minutes
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Proposed Answer: C

Explanation: 01-125MOV-100A, TRAIN A CLG VLV, is provided to allow air to be drawn through SBGT train A charcoal filter to remove heat produced from the decay of fission products. 01-125MOV-100A is interlocked with 01-125MOV-14A, TRAIN A INLET, as follows:

- When 01-125MOV-14A is opened, 01-125MOV-100A is automatically closed.
- When 01-125MOV-14A is closed, 01-125MOV-100A is automatically opened.

At time 55 minutes, SBGT train A is in service, therefore 01-125MOV-14A, TRAIN A INLET, is open. This results in 01-125MOV-100A being closed.

At time 65 minutes, SBGT train A is secured with 01-125MOV-14A closed. This results in 01-125MOV-100A being open.

A. Incorrect – At time 55 minutes, 01-125MOV-100A is closed.
B. Incorrect – At time 55 minutes, 01-125MOV-100A is closed. At time 65 minutes, 01-125MOV-100A is open.
D. Incorrect - At time 65 minutes, 01-125MOV-100A is open.

Technical Reference(s): OP-20, SDLP-01B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01B 1.05.b.6

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: 6/28 - Added “from 09-75” to stem condition based on validator comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A4.05
	Importance Rating	3.3

AC Electrical Distribution

Ability to manually operate and/or monitor in the control room: Voltage, current, power, and frequency on A.C. buses

Proposed Question: #47

The Plant is operating at 100% power with the following:

- Breaker 10514, the normal feeder breaker to Bus 10500, spuriously opens.
- EDG A starts and re-energizes Bus 10500.
- EDG C fails to start.
- Bus 10500 frequency is 62 Hz.

Which one of the following is the required action to restore Bus 10500 frequency to a normal value?

Rotate EDG A…

A. GOV switch in RAISE direction.

B. GOV switch in LOWER direction.

C. VOLT REG switch in RAISE direction.

D. VOLT REG switch in LOWER direction.

Proposed Answer: B

Explanation: The given Bus 10500 frequency is higher than the normal 60 Hz. Bus 10500 frequency is lowered by taking EDG A GOV switch in the lower direction.

A. Incorrect – This would raise frequency even higher.
C. Incorrect – This would raise voltage.
D. Incorrect – This would lower voltage.

Technical Reference(s): OP-22 Section G.9

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-93 1.05.a.1

Question Source: Modified Bank NMP1 2009 NRC #51

Question History: NMP1 2009 NRC #51

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 2.1.20
	Importance Rating	4.6

AC Electrical Distribution

Ability to interpret and execute procedure steps.

Proposed Question: #48

The Plant is operating at 100% power with the following:

- Current time is 1100 on a Tuesday.
- Main Generator real load is 873 MWe.
- Main Generator reactive load is 175 MVAR.
- Main Generator terminal voltage is 24.2 kV.
- Main Generator hydrogen gas pressure is 60 psig.
- Main Generator voltage regulator is in MAN.
- On-site and off-site electrical distribution systems are in normal lineups with the following:
    - Scriba substation breakers R100 and R935 are in service.
    - Backup protective relaying (“B package”) is in service.
    - Marcy Edic UE1 7 circuit is in service.
    - Scriba B bus is in service.

Which one of the following describes the status of Main Generator parameters?

Main Generator…

A. parameters are acceptable.

B. reactive load is too high for the given real load.

C. reactive load is too low for the given terminal voltage.

D. reactive load is too high for the given terminal voltage.

Proposed Answer: C

Explanation: OP-11A Section E.1 and associated attachments provide limitations on Main Generator parameters. With the voltage regulator in MAN and a normal electrical lineup, parameters must meet the limitations of attachments 2 and 9. The given values for real load, reactive load, and hydrogen gas pressure are within the limitations of attachment 2. However, the given values for real load, reactive load, and generator terminal voltage are outside of the required range for attachment 9. The voltage regulator must be adjusted to raise MVARs above the power line for 840-896 MWe operation.

A. Incorrect – Parameters are not within the limits of OP-11A attachment 9.
B. Incorrect – Reactive load and real load are within the limitations of OP-11A attachment 2.
D. Incorrect – Reactive load is too low for the given terminal voltage, not too high.

Technical Reference(s): OP-11A

Proposed references to be provided to applicants during examination: OP-11A section E.1, attachments 2-14

Learning Objective: SDLP-94D 1.13.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K3.08
	Importance Rating	2.7

UPS (AC/DC)

Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: Computer operation: Plant-Specific

Proposed Question: #49

The Plant is operating at 100% power when the following occurs:

- Electrical faults cause both L-25 and L-26 to de-energize.
- Both L-25 and L-26 will **NOT** be re-energized for an extended period of time due to damage.

Which one of the following describes the resulting operation of EPIC?

EPIC…

A. stops operating immediately.

B. continues to operate on battery power for approximately one hour.

C. continues to operate on battery power for approximately eight hours.

D. continues to operate on the Bypass Transformer indefinitely.

Proposed Answer: B

Explanation: EPIC receives power from a dedicated UPS. This UPS can receive power from either L-25 or L-26 through a manual transfer switch. Since both L-25 and L-26 are de-energized, the EPIC UPS has lost AC input power. The EPIC UPS has two dedicated batteries that are rated for approximately one hour. Upon loss of the AC input, the UPS immediately and automatically supplies EPIC with power from these batteries. When the batteries are depleted, EPIC loses power and ceases to operate, since the bypass transformer does not have power from L-25 or L-26.

A. Incorrect – The EPIC UPS batteries continue to supply power to EPIC for approximately one hour.
C. Incorrect – The EPIC UPS batteries are only rated for approximately one hour of operation following loss of L-25 and L-26. Four hours is the Station Blackout coping time.
D. Incorrect – The EPIC UPS batteries are only rated for approximately one hour of operation following loss of L-25 and L-26. Additionally, power will be unavailable from the bypass transformer due to loss of L-25 and L-26.

Technical Reference(s): OP-38C, SDLP-09A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-09A 1.10.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: Replaced 6/20 based on TR comment original was minutia

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 K1.04
	Importance Rating	2.6

DC Electrical Distribution

Knowledge of the physical connections and/or cause-effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: Ground detection

Proposed Question: #50

The Plant is shutdown for a refueling outage with the following:

- Annunciator 09-8-1-21, 125VDC BATT CHGR A DC GRD, is in alarm.
- 125 VDC BUS A GND DET meter on Panel 09-8 indicates +100 VDC.
- AOP-22, DC Power System A Ground Isolation, is entered.

Then, an Operator in the field opens a breaker for a Reactor Building emergency lighting panel.

- 125 VDC BUS A GND DET meter on Panel 09-8 now indicates -75 VDC.

Which one of the following describes the implications of these ground detection indications?

The given Reactor Building emergency lighting panel ___(1)___. Further ground isolation is ___(2)___.

	(1)	(2)
A.	is free of grounds	**NOT** required
B.	is free of grounds	required
C.	has a ground	**NOT** required
D.	has a ground	required

Proposed Answer: D

Explanation: AOP-22 requires ground isolation when the ground is greater than 50 VDC, regardless of polarity. Based on the indication of ground detection changing a net 175 VDC, the given Reactor Building emergency lighting panel has a significant ground. However, the subsequent -75 VDC ground indication means another ground is present on the DC system, requiring further ground isolation activities.

A. Incorrect – The emergency lighting has a ground. The remaining -75 VDC ground indication requires further ground isolation activities.
B. Incorrect – The emergency lighting has a ground.
C. Incorrect – The remaining -75 VDC ground indication requires further ground isolation activities.

Technical Reference(s): AOP-22

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.01, 1.03, 1.10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K3.01
	Importance Rating	4.2

EDGs

Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: Emergency core cooling systems

Proposed Question: #51

The Plant was operating at 100% power when a Loss of Coolant Accident and Loss of Offsite Power resulted in the following:

Time (seconds)	Event
0	Drywell pressure is 2.7 psig and rising. Reactor water level is 120” and lowering. Lines 3 and 4 de-energize.
10	Only one Emergency Diesel Generator (EDG) loads onto its respective emergency bus. The other three EDGs have locked out due to generator faults.
35	Reactor pressure is 220 psig and lowering.

Which one of the following describes the number of Residual Heat Removal (RHR) and Core Spray (CS) Pumps injecting to the Reactor at time 45 seconds?

	Number of RHR Pumps Injecting	Number of Core Spray Pumps Injecting
A.	Two	One
B.	Two	Zero
C.	One	One
D.	One	Zero

Proposed Answer: C

Explanation: All four EDGs have a start signal due to both high drywell pressure and undervoltage. The undervoltage signal also gives the EDGs a signal to load onto the emergency buses. With only one EDG loaded, one emergency bus is de-energized, resulting in two of four RHR Pumps and one of two CS Pumps being unavailable. Additionally, one additional RHR Pump on the energized emergency bus will not start due to lack of two EDGs tied to the bus. Therefore only one RHR Pump and one CS Pump have started. With Reactor pressure less than 450 psig, both systems will be injecting. The EDGs and ECCS systems are designed to ensure injection can be established before 45 seconds have elapsed.

Technical Reference(s): SDLP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14 1.10.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: changed time 35 rpv pressure to 250 based on validator comment 7/3/DK changed to 220 psig based on val com

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K1.04
	Importance Rating	2.8

Instrument Air

Knowledge of the connections and / or cause-effect relationships between INSTRUMENT AIR SYSTEM and the following: Cooling water to compressor

Proposed Question: #52

The Plant is shutdown for a refueling outage with the following:

- Only Air Compressor A is running.
- Air Compressor B and C control switches are in Pull-to-Lock to support maintenance.
- Air Compressor B is being started for post-maintenance testing.

Which one of the following describes the source of cooling water to Air Compressor B?

The cooling water for Air Compressor B is from ___(1)___. The cooling water flow to Air Compressor B is ___(2)___.

	(1)	(2)
A.	RBCLC	manually established prior to Air Compressor B start.
B.	RBCLC	automatically initiated when Air Compressor B starts.
C.	TBCLC	manually established prior to Air Compressor B start.
D.	TBCLC	automatically initiated when Air Compressor B starts.

Proposed Answer: D

Explanation: Air Compressors are cooled by TBCLC. TBCLC flow is controlled by an automatic solenoid that opens on compressor start. With Air Compressor B secured, the solenoid is closed. When Air Compressor B starts, the solenoid opens. This results in more total TBCLC flow.

A. Incorrect – Air Compressors are cooled by TBCLC, not RBCLC. Total TBCLC flow rate rises upon Air Compressor start due to automatic opening of solenoid valve in cooling water supply line.
B. Incorrect – Air Compressors are cooled by TBCLC, not RBCLC.
C. Incorrect – Total TBCLC flow rate rises upon Air Compressor start due to automatic opening of solenoid valve in cooling water supply line.

Technical Reference(s): OP-39, SDLP-39

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.10.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments: Reworded second half of question for technical accuracy based on validator comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 K6.04
	Importance Rating	3.0

Component Cooling Water

Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Pumps

Proposed Question: #53

The plant is operating at 100% power with the following:

- Lake water temperature is 76^{o}F.
- Service Water Pumps A and B are in service.
- Service Water Pump C is in standby.
- Service Water discharge header pressure is 95 psig.
- Then, Service Water Pump A breaker trips on overcurrent.
- Service Water discharge header pressure lowers to 80 psig.

Which one of the following describes the status of Service Water Pump C?

Service Water Pump C…

A. auto-starts due to low discharge header pressure.

B. auto-starts due to Service Water Pump A breaker trip.

C. remains in standby unless discharge header pressure lowers further.

D. remains in standby until Operators manually start it.

Proposed Answer: B

Explanation: Service Water Pump C will auto-start if either another Service Water Pump breaker trips or header pressure lowers below 75 psig. Therefore, Service Water Pump C will auto-start due to Service Water Pump A breaker trip.

A. Incorrect – Pressure has not lowered to 75 psig.
C. Incorrect – Service Water Pump C will auto-start due to Service Water Pump A breaker trip.
D. Incorrect – Service Water Pump C will auto-start due to Service Water Pump A breaker trip.

Technical Reference(s): OP-42, ESK-5AF

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-46A 1.05.c.2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments: 6/28/12 - Revised question and answers to make easier, based on validation comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201001 K4.12
	Importance Rating	2.9

CRD Hydraulic

Knowledge of CONTROL ROD DRIVE HYDRAULIC SYSTEM design feature(s) and/or interlocks which provide for the following: Controlling CRD system flow

Proposed Question: #54

The plant is operating at 100% power with the in-service Control Rod Drive (CRD) Flow Control Valve in automatic when a Reactor scram occurs.

Which one of the following describes the CRD Flow Control Valve response and the CRD total system flow change?

	CRD Flow Control Valve	CRD Total System Flow
A.	Closes	Rises
B.	Closes	Lowers
C.	Opens	Rises
D.	Opens	Lowers

Proposed Answer: A

Explanation: During normal operation, the CRD Flow Control Valve automatically maintains CRD total system flow at approximately 60 gpm. During a scram, CRD total system flow rises significantly due to flow through all 137 scram inlet valves. This flow is sensed through CRD system flow element FE-203 and transmitted to the Flow Control Valve, which closes in an attempt to limit total system flow below setpoint.

B. Incorrect – Total system flow rises due to flow to all 137 scram inlet valves.
C. Incorrect – The CRD Flow Control Valve closes in response to rising system flow.
D. Incorrect – The CRD Flow Control Valve closes in response to rising system flow. Total system flow rises due to flow to all 137 scram inlet valves.

Technical Reference(s): OP-25, FM-27A, SDLP-03C

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C 1.05.c.3

Question Source: Modified Bank NMP1 2008 NRC #36

Question History: NMP1 2008 NRC #36

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201006 K5.13
	Importance Rating	3.5

RWM

Knowledge of the operational implications of the following concepts as they apply to ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC): Insert block: P-Spec (Not-BWR6)

Proposed Question: #55

A Plant startup is in progress with the following:

- Reactor power is 5% on APRMs.
- The Rod Worth Minimizer (RWM) NORMAL BYPASS keylock switch is in NORMAL.
- An Operator is withdrawing the first control rod (22-31) in a group.
- The group withdraw limit is 24.
- The Operator erroneously withdraws control rod 22-31 to position 26.
- Then, the Operator selects the next control rod in the same group (30-31).

Which one of the following describes the ability to move control rods 30-31 and 22-31 with the Reactor Manual Control System (RMCS)?

	Control Rod 30-31…	Control Rod 22-31…
A.	can NOT be moved.	can be inserted only.
B.	can be withdrawn and inserted.	can be inserted only.
C.	can NOT be moved.	can be withdrawn and inserted.
D.	can be withdrawn and inserted.	can be withdrawn and inserted.

Proposed Answer: A

Explanation: With Reactor power below 10% and the RWM NORMAL BYPASS keylock switch in NORMAL, the RWM is enforcing the programmed rod sequence. Control rod 22-31 has been withdrawn beyond the withdraw limit. This results in a withdraw error and withdraw block for control rod 22-31. Control rod 22-31 can be inserted with RMCS to correct the error. Until the withdraw error is corrected, all other control rods have withdraw and insert blocks imposed.

B. Incorrect – Control rod 30-31 withdraw and insert are blocked.
C. Incorrect – Control rod 22-31 withdraw is blocked.
D. Incorrect – Control rod 22-31 withdraw is blocked. Control rod 30-31 withdraw and insert are blocked.

Technical Reference(s): OP-64

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03D 1.05.b.4, 1.05.b.5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202002 A3.02
	Importance Rating	3.4

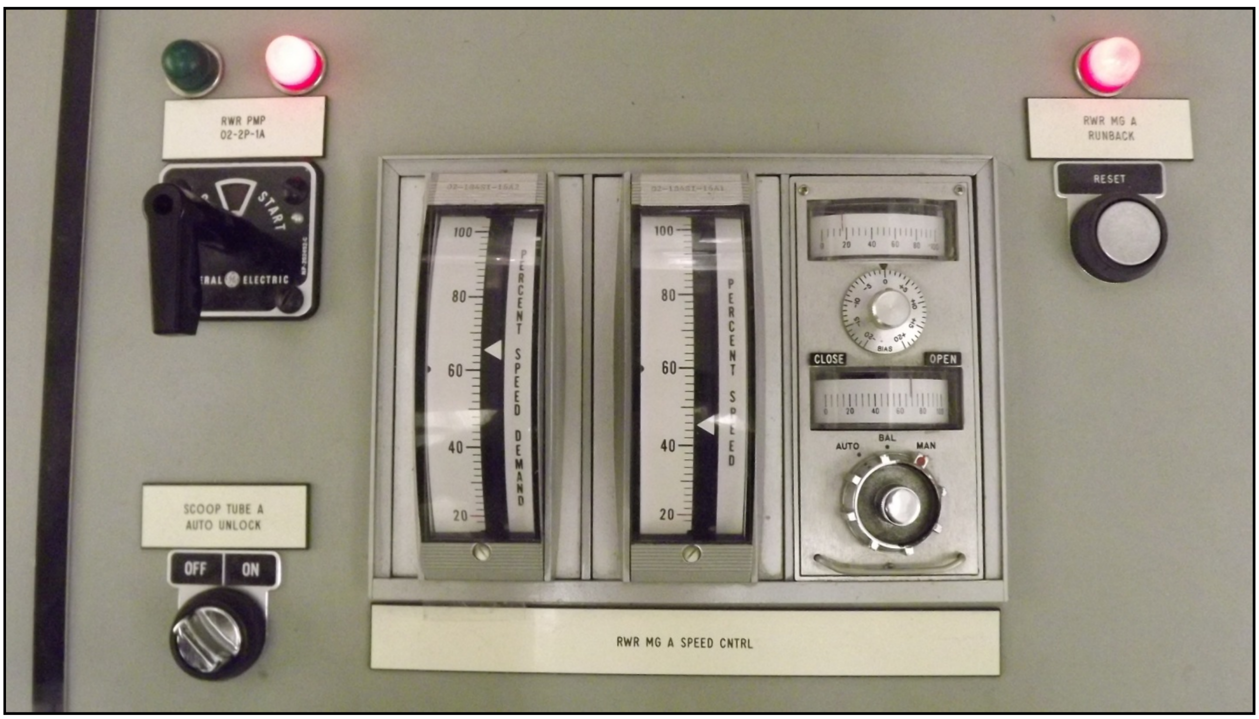
Recirculation Flow Control

Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: Lights and alarms

Proposed Question: #56

The Plant is operating at 65% power with the following:

- A loss of all Panel 09-5 annunciators occurs.
- Then, during monitoring of Panel 09-5, the following stable indications are observed for Reactor Water Recirculation (RWR) Pump A:



Which one of the following describes the indicated condition of RWR Pump A?

RWR Pump A…

A. has tripped.
B. has received a 30% runback signal.
C. has received a 44% runback signal.
D. is in a normal condition for 65% power operation.

Proposed Answer: C

Explanation: The given indications show RWR Pump A control switch red flagged with red light on and green light off. This indicates RWR Pump A is running. The RWR Pump A speed controller shows a PERCENT SPEED of 44% with the RWR MG A RUNBACK red light on. This indicates a 44% runback has occurred. Normal indications for 65% power would have the RWR MG A RUNBACK red light off.

A. Incorrect – RWR Pump A control switch is red flagged with red light on and green light off, indicating the pump has not tripped.
B. Incorrect – PERCENT SPEED DEMAND is 44%, indicating this is a 44% runback, not a 30% runback.
D. Incorrect – Normal indications for 65% power would have the RWR MG A RUNBACK red light off.

Technical Reference(s): OP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I 1.05.b.2, 1.12.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	215001 K4.01
	Importance Rating	3.4

Traversing In-core Probe

Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation: Mark-I&II (Not-BWR1)

Proposed Question: #57

The Plant is operating at 100% power with the following:

- Traversing In-Core Probe (TIP) scans are in progress with TIP System 1 detector in the core.
- A coolant leak develops in the Drywell.
- Drywell pressure is 3 psig and rising.
- TIP System 1 detector fails to retract automatically and manually.

Which one of the following describes the resulting operation of the TIP System 1 shear valve?

The TIP System 1 shear valve…

A. automatically closes immediately.

B. automatically closes after a time delay.

C. must be manually closed using a switch at the TIP Room.

D. must be manually closed using a switch on Control Room Panel 09-13.

Proposed Answer: D

Explanation: The TIP shear valves are provided as a backup to the normal primary containment isolation feature. The normal primary containment isolation feature automatically retracts the detector and then closes the associated ball valve. If the detector fails to retract, the ball valve cannot close, thus the shear valve is provided to cut through the detector cable and isolate the containment penetration. No automatic shear valve actuation is provided. The operator must detect the failure and manually actuate the shear valve. The control switch for this action is on Control Room Panel 09-13.

A. Incorrect – The shear valve does NOT automatically close.
B. Incorrect – The shear valve does NOT automatically close.
C. Incorrect – The shear valve control is located on Control Room Panel 09-13.

Technical Reference(s): AOP-15, SDLP-07F

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07F 1.05.b.1, 1.05.b.2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	216000 K1.09
	Importance Rating	3.7

Nuclear Boiler Inst.

Knowledge of the physical connections and/or cause-effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following: Redundant reactivity control/ alternate rod insertion: Plant-Specific

Proposed Question: #58

The Plant is operating at 100% power when a transient results in the following:

- Reactor pressure reaches a peak of 1115 psig and then lowers back to 970 psig.
- Reactor water level reaches a low of 100” and then rises back to 200”.

Which one of the following describes the status of the Alternate Rod Insertion (ARI) System?

ARI has…

A. **NOT** initiated.

B. initiated. The initiation logic was satisfied by Reactor pressure only.

C. initiated. The initiation logic was satisfied by Reactor water level only.

D. initiated. The initiation logic was satisfied by Reactor pressure AND water level.

Proposed Answer: C

Explanation: ARI can be initiated by either Reactor Lo-Lo water level (110” setpoint, 105.4” TS allowable value) or Reactor Hi pressure (1139 psig setpoint, 1153 psig TS allowable value). With the given conditions, Reactor water level has gone below the ARI setpoint, but Reactor pressure has remained below the ARI setpoint. Therefore ARI has initiated due to Reactor water level alone.

A. Incorrect – ARI has initiated due to Reactor water level.
B. Incorrect – Reactor pressure has NOT exceeded the ARI setpoint of 1139 psig.
D. Incorrect – Reactor pressure has NOT exceeded the ARI setpoint of 1139 psig.

Technical Reference(s): OP-25, ARP-09-5-1-35, ESK-7FA, ESK-7FB

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP- 1.05.c.8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	223001 K1.15
	Importance Rating	3.5

Primary CTMT and Aux.

Knowledge of the physical connections and/or cause-effect relationships between PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES and the following: HPCI: Plant-Specific

Proposed Question: #59

The Plant was operating at 100% power when a seismic event resulted in the following:

- Drywell pressure is 4 psig and rising slowly.
- Reactor water level is 200" and stable with HPCI injecting.
- Condensate Storage Tank (CST) water level is 90".
- Torus water level is 15'.

Which one of the following describes the HPCI Pump suction source?

The HPCI Pump is taking a suction from…

A. the CSTs because this is the normally aligned source.

B. the Torus because this is the normally aligned source.

C. the Torus because Torus water level is high.

D. the Torus because CST water level is low.

Proposed Answer: C

Explanation: The HPCI Pump suction is normally aligned to the CST. The alternate HPCI Pump suction supply is the Torus. The HPCI Pump suction supply shifts from the CSTs to the Torus when any of the following occur:

- Low CST water level – less than 59.5”
- CST manual suction isolation valves NOT full open
- High Torus water level – greater than 14.5’

Since Torus water level is above 14.5’, the HPCI Pump suction has transferred to the Torus.

A. Incorrect – HPCI Pump suction automatically transfers to the Torus when Torus water level rises to 14.5’.
B. Incorrect – The Torus is NOT the normally aligned suction source for the HPCI Pump.
D. Incorrect – CST water level is NOT below the setpoint for automatic transfer of HPCI Pump suction.

Technical Reference(s): OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.05.b.1

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	226001 K2.02
	Importance Rating	2.9

RHR/LPCI: CTMT Spray Mode

Knowledge of electrical power supplies to the following: Pumps

Proposed Question: #60

The Plant was operating at 100% power when a manual scram was inserted due to a steam leak in the Drywell with the following:

- Both 115 KV power lines de-energize.
- Emergency Diesel Generators (EDGs) A and C fail to start.
- Reactor water level is 200” and stable with only RCIC injecting.
- Torus Spray is required by EOP-4, Primary Containment Control.

Which one of the following lists the Residual Heat Removal (RHR) Pump that is currently available for Torus Spray?

A. A

B. B

C. C

D. D

Proposed Answer: D

Explanation: The combination of a manual scram, loss of 115 KV power lines, and failure of EDGs A and C results in Bus 10500 being de-energized and Bus 10600 being energized. Bus 10500 is the power supply to RHR Pumps A and B. Bus 10600 is the power supply to RHR Pumps C and D. Therefore, only RHR Pumps C and D have power. However, RHR Pump C sprays the Torus through 10MOV-39A and 10MOV-38A. Both of these valves have no power due to the loss of Bus 10500. Therefore, only RHR Pump D is capable of spraying the Torus.

A. Incorrect – RHR Pump A is unavailable due to loss of power to Bus 10500.
B. Incorrect – RHR Pump B is unavailable due to loss of power to Bus 10500.
C. Incorrect – RHR Pump C is unavailable for Torus Spray since 10MOV-39A and 10MOV-38A have no power.

Technical Reference(s): OP-22, OP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.03, 1.10.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: DK\TH 4/23: enhanced ‘question’ part by adding “currently” based on NRC review.

TRH 6/21 - Changed answers since RHR Pump C is unavailable for Torus Spray due to loss of power to 10MOV-39A and 10MOV-38A, based on validator comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	233000 A2.09
	Importance Rating	2.7

Fuel Pool Cooling/Cleanup

Ability to (a) predict the impacts of the following on the FUEL POOL COOLING AND CLEAN-UP; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical power failures

Proposed Question: #61

The Plant is operating at 100% power with the following:

- Fuel Pool Cooling (FPC) Recirc Pump A is in service.
- FPC Recirc Pump B is in standby.

Then, MCC 131 de-energizes due to an electrical fault.

Which one of the following describes the impact of this electrical failure on the FPC System and the corresponding Operator action, if any?

A. The standby FPC Recirc Pump is unavailable. No Operator action is required for the FPC System.

B. The running FPC Recirc Pump trips. Verify the standby FPC Recirc Pump auto-started per AOP-68, Spent Fuel Pool Trouble.

C. The running FPC Recirc Pump trips. Restore FPC using the standby FPC Recirc Pump per AOP-68, Spent Fuel Pool Trouble.

D. The running FPC Recirc Pump trips and the standby FPC Recirc Pump is unavailable. Restore FPC using the Decay Heat Removal (DHR) System per OP-30B, Decay Heat Removal System.

Proposed Answer: C

Explanation: FPC Recirc Pump A is powered from MCC 131 and FPC Recirc Pump B is powered from MCC 141. Since FPC Recirc Pump A was running, the loss of MCC 131 results in trip of the running FPC Recirc Pump. FPC Recirc Pump B still has electrical power available from MCC 141, but does not have an auto-start feature. AOP-68, Spent Fuel Pool Trouble, directs restoration of FPC using the standby pump.

A. Incorrect – The standby FPC Recirc Pump (B) is available with power from MCC 141.
B. Incorrect – The standby FPC Recirc Pump (B) does not auto-start on trip of the running pump.
D. Incorrect – The standby FPC Recirc Pump (B) is available with power from MCC 141.

Technical Reference(s): OP-30, AOP-68, OP-30B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-19 1.10.b, 1.15

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	2
K/A #	259001 K5.03
Importance Rating	2.8

Reactor Feedwater

Knowledge of the operational implications of the following concepts as they apply to REACTOR FEEDWATER SYSTEM: Turbine operation: TDRFPs-Only

Proposed Question: #62

Which one of the following describes the upper limitation on Reactor Feed Pump Turbine (RFPT) steady-state operating speed, per OP-2A, Feedwater System?

RFPT operation greater than ___(1)___ rpm should be minimized to ___(2)___.

	(1)	(2)
A.	4300	reduce bucket vibratory stress
B.	4300	prevent loss of MGU speed control
C.	5500	reduce bucket vibratory stress
D.	5500	prevent loss of MGU speed control

Proposed Answer: A

Explanation: OP-2A Precaution C.2.3 states, “Steady state operation of RFPT greater than 4300 rpm should be minimized to reduce bucket vibratory stresses.”

B. Incorrect – RFPT operation should be maintained **above 1000 rpm** to prevent loss of MGU speed control.
C. Incorrect – 4300 rpm is the steady-state operating limitation. 5500 rpm is the overspeed trip setpoint.
D. Incorrect – 4300 rpm is the steady-state operating limitation. 5500 rpm is the overspeed trip setpoint. RFPT operation should be maintained **above 1000 rpm** to prevent loss of MGU speed control.

Technical Reference(s): OP-2A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-33 1.05.a.3.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments: 7/3/12 - Changed question due to high validation miss rate.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	272000 A3.08
	Importance Rating	2.9

Radiation Monitoring

Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including: Meter indications

Proposed Question: #63

A Plant startup is in progress with the following:

- A Condenser Air Removal Pump is in service to establish Main Condenser vacuum.
- Reactor pressure is 150 psig and slowly rising.
- One Turbine Bypass Valve is partially open.

Then, Annunciator 09-3-3-1, MAIN STM RAD MON HI, is received and followed shortly by Annunciator 09-5-1-32, MAIN STM LINE RADIATION HI-HI.

Main Steam Line radiation monitor indications are as shown **(see next page).**

Which one of the following describes the plant response to these indications?

A. No automatic plant response has occurred yet.

B. The Condenser Air Removal Pump has tripped. The MSIVs remain open.

C. The MSIVs have closed. The Condenser Air Removal Pump remains in service.

D. The Condenser Air Removal Pump has tripped AND the MSIVs have closed.

MAIN STM RAD MON A

MAIN STM RAD MON C

MAIN STM RAD MON B

MAIN STM RAD MON D

Proposed Answer: A

Explanation: Four radiation monitors (A, B, C, D) are provided on the Main Steam Lines. A and C input to trip system A. B and D input to trip system B. The trip logic requires at least one of the trip system A inputs and at least one of the trip system B inputs to be above the Hi-Hi setpoint (2871 mr/hr nominal) to cause the associated MSL rad monitor trips. These trips include:

- Recirc Loop Sample Valves 02-2AOV-39 and 02-2AOV-40 close
- Main Steam Line Drain Valve 29MOV-74 and 29MOV-77 close
- Condenser Air Removal Pumps 38P-2A and 38P-2B trip
- Condenser Air Removal Pump Suction and Discharge Isolation Valves 38AOV-111 and 38AOV-112 close

The given MSL rad monitor indications show 3 monitors (A, C, and D) above the Hi setpoint (1809 mr/hr nominal) but below the Hi-Hi setpoint (2871 mr/hr nominal), and 1 monitor (B) above the Hi-Hi setpoint. With only one monitor above the Hi-Hi setpoint, the trip logic is NOT satisfied, therefore the Condenser Air Removal Pump has NOT tripped. The MSIVs do NOT go closed on high MSL rad level. This automatic isolation was removed by previous plant modification.

Technical Reference(s): ARP 09-3-3-1, ARP-09-5-1-32, OP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-17 1.05.a.3, 1.05.c.5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290003 A4.01
	Importance Rating	3.2

Control Room HVAC

Ability to manually operate and/or monitor in the control room: Initiate/reset system

Proposed Question: #64

The Plant is operating at 100% power with the following:

- A Loss of Coolant Accident (LOCA) at Nine Mile Point Unit 2 is causing high radiation levels on-site at JAF.
- The Control Room Ventilation Supply radiation monitor is reading 100 cpm and rising.
- Annunciator 09-75-1-20, CNTRL RM SUPP RAD MON INOP OR HI, is clear.

Which one of the following describes the radiation level at which Annunciator 09-75-1-20 will alarm and the required response upon receipt of the alarm?

Annunciator 09-75-1-20 will alarm when the Control Room Ventilation Supply radiation monitor reaches ___(1)___ cpm. Then, the Control Room Ventilation System ___(2)___.

	(1)	(2)
A.	200	automatically goes into the ISOLATE mode due to the high radiation signal.
B.	200	must be manually placed in the ISOLATE mode using the ISOL & PURGE CNTRL switch on Panel 09-75.
C.	1000	automatically goes into the ISOLATE mode due to the high radiation signal.
D.	1000	must be manually placed in the ISOLATE mode using the ISOL & PURGE CNTRL switch on Panel 09-75.

Proposed Answer: D

Explanation: Annunciator 09-75-1-20 alarms on high Control Room Ventilation Supply radiation level at 1000 cpm. This alarm requires placing Control Room Ventilation in the ISOLATE mode as soon as practicable and in all cases within 30 minutes. However, this alarm does NOT automatically place Control Room Ventilation in the ISOLATE mode. OP-55B gives the operator direction on how to place the system in the ISOLATE mode. This is accomplished by placing the ISOL & PURGE CONTRL switch in ISOL. From this one switch manipulation, the system enters the ISOLATE mode.

A. Incorrect – The high radiation alarm setpoint is 1000 cpm. The system does NOT automatically enter the ISOLATE mode upon alarm.
B. Incorrect – The high radiation alarm setpoint is 1000 cpm.
C. Incorrect – The system does NOT automatically enter the ISOLATE mode upon alarm.

Technical Reference(s): ARP 09-75-1-20, OP-55B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-70 1.07.a and 1.14

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290002 A2.05
	Importance Rating	3.7

Reactor Vessel Internals

Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: †Exceeding thermal limits

Proposed Question: #65

The Plant is operating at 95% power during a startup when the Average Planar Linear Heat Generation Rate (APLHGR) is found to be exceeded.

Which one of the following describes the significance of the thermal limit violation on the nuclear fuel and the required action?

This thermal limit violation signifies that the nuclear fuel may experience __(1)__. A manual Reactor scram __(2)__.

	(1)	(2)
A.	inadequate cooling during a Loss of Coolant Accident	is immediately required
B.	inadequate cooling during a Loss of Coolant Accident	is **NOT** immediately required
C.	onset of transition boiling during transient operation	is immediately required
D.	onset of transition boiling during transient operation	is **NOT** immediately required

Proposed Answer: B

Explanation: APLHGR is established to ensure that no fuel damage occurs during a loss of coolant accident. In the event APLHGR is violated, Technical Specifications require restoration of APLHGR to within limits within 2 hours (such as by a power reduction or control rod pattern adjustment), but do **NOT** require an immediate reactor scram.

A. Incorrect – In the event of thermal limit violation, Technical Specifications allow 2 hours to restore the thermal limit, and then 4 more hours to reduce core thermal power below 25%.
C. Incorrect – Violation of MCPR may result in OTB during transient conditions. In the event of thermal limit violation, Technical Specifications allow 2 hours to restore the thermal limit, and then 4 more hours to reduce core thermal power below 25%.
D. Incorrect – Violation of MCPR may result in OTB during transient conditions.

Technical Reference(s): Tech Spec 3.2.2, Tech Spec Bases 3.2.2

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-LOI-Therm09 18, SDLP-02G 1.16

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments: Changed to APLHGR based on validator comment

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.37
	Importance Rating	4.3

Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Proposed Question: #66

Which one of the following identifies manipulations that are acceptable per the reactivity management guidelines of EN-OP-115, Conduct of Operations?

(1) Operating RWR MG A SPEED CNTRL and RWR MG B SPEED CNTRL manual adjustment knobs simultaneously to raise Reactor power.

(2) Operating RWR MG A SPEED CNTRL and RWR MG B SPEED CNTRL manual adjustment knobs simultaneously during a rapid power reduction.

(3) Operating ROD EMERG IN NOTCH OVERRIDE and ROD MOVEMENT CNTRL switches simultaneously to continuously withdraw a control rod.

(4) Operating ROD EMERG IN NOTCH OVERRIDE switch and Rod Select pushbuttons simultaneously to insert control rods during a failure to scram.

A. (1) and (2) only

B. (3) and (4) only

C. (1), (2), and (3) only

D. (2), (3), and (4) only

Proposed Answer: D

Explanation: EN-OP-115 section 5.5 contains reactivity management guidance. Step [9] states, “Positive reactivity additions are performed using one method at a time unless, for PWRs only, in accordance with an approved procedure or reactivity plan.” EN-OP-115 attachment 9.5 section 3 gives a JAF-specific list of acceptable two-handed operations, including some reactivity manipulations. The following are included:

- Rapid power reduction with RWR.
- Inserting control rods per EP-3 using RMCS where inserting a control rod with the “Rod Emerg In Notch Override” switch is performed as a static activity while using the other hand to perform activities such as ranging down/up on IRMs, adjusting drive DP, selecting rods, etc.
- Actions that are procedurally directed or cannot physically be performed using one hand.

OP-26, Control Rod Drive Manual Control System, section E.2 contains guidance to operate ROD EMERG IN NOTCH OVERRIDE and ROD MOVEMENT CNTRL switches simultaneously to continuously withdraw a control rod.

OP-27 does NOT have any allowance for operating RWR MG A SPEED CNTRL and RWR MG B SPEED CNTRL manual adjustment knobs simultaneously to **raise** Reactor power.

A. Incorrect – (1) is NOT allowed, while (3) and (4) are allowed.
B. Incorrect – (2) is also allowed.
C. Incorrect – (1) is NOT allowed, while (4) is allowed.

Technical Reference(s): EN-OP-115, OP-26, OP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-Admin 1.02

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: enhanced B as to what kind of power reduction per validator comment

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.1
	Importance Rating	3.8

Knowledge of conduct of operations requirements.

Proposed Question: #67

The Plant is operating at 100% power with the following:

- You are the At-The-Controls (ATC) Operator.
- You are being relieved by another qualified Reactor Operator for lunch.

Which one of the following describes the requirements that must be met for this watch station relief per EN-OP-115-03, Shift Turnover and Relief?

(1) Verbal turnover to on-coming Reactor Operator.
(2) Update brief informing the shift crew of the relief change.
(3) Permission from the Shift Manager (SM) or Control Room Supervisor (CRS).

A. (1) only.

B. (1) and (2) only.

C. (1) and (3) only.

D. (1), (2), and (3).

Proposed Answer: D

Explanation: EN-OP-115-03 requires the following for a control room operator to be relieved during their shift:

- Permission granted by the SM or CRS as applicable.
- A verbal turnover conducted to a qualified individual as follows…
- An update brief performed informing the shift crew of the relief change.

A. Incorrect – An update brief and SM/CRS permission are also required.
B. Incorrect – SM/CRS permission is also required.
C. Incorrect – An update brief is also required.

Technical Reference(s): EN-OP-115-03

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-Admin 1.02

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.12
	Importance Rating	3.7

Knowledge of surveillance procedures.

Proposed Question: #68

The Plant is shutdown with the following:

- ST-1S, Shutdown Cooling Containment Isolation Valve Testing (IST), is in progress.
- 10MOV-17, SHUTDOWN CLG SUCT, closing time has been recorded as 24.2 sec.
- ST-1S contains the following acceptance criteria:

Level 1 Acceptance Criteria

CLOSING TIMES		
Valve	IST Stroke Time (sec)	Step
10MOV-18	23.2 to 38.0	8.1.6 or 8.2.8
10MOV-17	12.9 to 21.6	8.1.10 or 8.2.4

Level 2 Acceptance Criteria

CLOSING TIMES		
Valve	IST Stroke Time (sec)	Step
10MOV-18	26.3 to 35.6	8.1.6 or 8.2.8
10MOV-17	14.7 to 19.8	8.1.10 or 8.2.4

Which one of the following describes the test results and required actions?

A. Immediately retest the valve. If the second closing time is in the proper range, the test is satisfactory with **NO** corrective actions required.

B. Immediately retest the valve. If the second closing time is in the proper range, the test is satisfactory with corrective actions required.

C. An immediate retest is **NOT** allowed in this case. The test is **NOT** satisfactory.

D. An immediate retest is **NOT** required in this case. The test is satisfactory.

Proposed Answer: C

Explanation: The given stroke time of 24.2 seconds is above the maximum stroke time of 21.6 seconds allowed for valve 10MOV-17. This represents a failure to meet both Level 1 and 2 acceptance criteria. For failure to meet Level 1 acceptance criteria, ST-1S requires the test to be declared unsatisfactory and does not allow immediate retest to obtain a satisfactory stroke time.

A. Incorrect – Since the valve failed Level 1 acceptance criteria, no immediate retest is allowed.
B. Incorrect – Since the valve failed Level 1 acceptance criteria, no immediate retest is allowed.
D. Incorrect – The valve did not meet all acceptance criteria.

Technical Reference(s): ST-1S, AP-19.01

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16C 1.13.a

Question Source: Modified NMP1 2008 NRC #75

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.43
	Importance Rating	3.0

Knowledge of the process used to track inoperable alarms.

Proposed Question: #69

The Plant is operating at 100% power with the following:

- I&C is performing troubleshooting on an SDIV level instrument.
- Annunciator 09-5-1-44, SDIV A OR B NOT DRAINED, will be received intermittently for the next four (4) hours.
- The troubleshooting activity will be completed by the end of shift.

Which one of the following describes requirements for this annunciator per EN-OP-115-08, Annunciator Response?

The annunciator…

A. must be flagged and logged in either the annunciator log or turnover sheet.

B. must be flagged, but does **NOT** need to be logged in either the annunciator log or turnover sheet.

C. must be logged in either the annunciator log or turnover sheet, but does **NOT** need to be flagged.

D. does **NOT** need to be flagged nor logged in either the annunciator log or the turnover sheet.

Proposed Answer: B

Explanation: EN-OP-115-08 Section 5.2[12] describes requirements for flagging of annunciators. For short duration alarms (less than 15 minutes), the CRS/SM may waive flagging requirements. For activities that cause an expected alarm and the annunciator is associated with equipment required in current mode, the following is required:

- Install an annunciator flag on the expected alarm.
- If projected activity duration will exceed remaining time of current shift, then update annunciator log or SRO/RO turnover sheet as appropriate with alarm status.

A. Incorrect – Since the activity will be complete by end of shift, the annunciator does not have to be logged.
C. Incorrect – The annunciator must be flagged since the activity will be more than 15 minutes, but does not have to be logged due to completion by end of shift.
D. Incorrect – The annunciator must be flagged since the activity will be more than 15 minutes.

Technical Reference(s): EN-OP-115-08

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-Admin 1.02

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.35
	Importance Rating	3.6

Ability to determine Technical Specification Mode of Operation.

Proposed Question: #70

A Plant shutdown is in progress with the following:

- Control rod insertion is in progress.
- Reactor power is:
  - 2% on APRMs.
  - Mid-scale on IRM Range 9.
- Reactor pressure is 975 psig.
- The Mode switch is in START & HOT STBY.

Which one of the following is the Technical Specification Mode of Operation?

A. 1 (Power Operation)

B. 2 (Startup)

C. 3 (Hot Shutdown)

D. 4 (Cold Shutdown)

Proposed Answer: B

Explanation: Technical Specification Table 1.1-1 contains the criteria for Reactor Mode of Operation (see below). With the Mode switch in START & HOT STBY, the Reactor is in Mode 2, Startup.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel(a) or Startup/Hot Standby	NA
3	Hot Shutdown(a)	Shutdown	> 212
4	Cold Shutdown(a)	Shutdown	≤ 212
5	Refueling(b)	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

A. Incorrect – While the Reactor is still generating power, as evidenced by APRM reading, the Mode switch is NOT in RUN.
C. Incorrect – While a Plant shutdown is in progress, the Mode switch is NOT in SHUTDOWN.
D. Incorrect – While a Plant shutdown is in progress, the Mode switch is NOT in SHUTDOWN. Additionally, even though Reactor pressure is lower than the normal operating value, it is still higher than a pressure corresponding to 212^{o}F.

Technical Reference(s): Technical Specification Table 1.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-ITS02 1.03

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.11
	Importance Rating	3.8

Ability to control radiation releases.

Proposed Question: #71

The Plant was operating at 100% power when the following occurred:

- A Main Steam leak began discharging into the Turbine Building.
- The Reactor was scrammed.
- The MSIVs are stuck in mid-position.
- The running Turbine Building Exhaust Fan has tripped.
- The standby Turbine Building Exhaust Fan has failed to auto-start.

Which one of the following describes the action to take with Turbine Building Ventilation per EOP-6, Radioactivity Release Control, and why?

A. Restore a Turbine Building Exhaust Fan to service to limit unmonitored ground level radioactivity release.

B. Restore a Turbine Building Exhaust Fan to service to prevent equipment damage in the Turbine Building.

C. Ensure Turbine Building Ventilation is secured and isolated to contain the steam leak to the Turbine Building.

D. Ensure Turbine Building Ventilation is secured and isolated to limit total radioactivity release to the Site Boundary.

Proposed Answer: A

Explanation: EOP-6 has a step that states, “IF Turbine Building Ventilation or Radwaste Building Ventilation is shutdown, or isolated due to high radiation, THEN Restart the ventilation system as required. Defeat isolation interlocks if necessary (EP-2).” The restart of Turbine Building Ventilation is required to direct any radioactive discharge to an elevated, monitored release point instead of a ground-level, unmonitored release point.

B. Incorrect – The reason for restarting Turbine Building Ventilation is to control radioactive release, not prevent equipment inside the Turbine Building.
C. Incorrect – EOP-6 requires Turbine Building Ventilation to be restarted, not isolated.
D. Incorrect – EOP-6 requires Turbine Building Ventilation to be restarted, not isolated.

Technical Reference(s): EOP-6, MIT-301.11G

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11G 6.04

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.7
	Importance Rating	3.5

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Proposed Question: #72

The Plant is operating at 100% power with the following:

- Traversing In-Core Probe (TIP) scans are in progress.
- The detector position indication on TIP System 1 is not working properly.
- The TIP detector is believed to be in the shield.
- An Operator is to be sent into the TIP Room to verify the detector position.
- The TIP Room is posted as a Very High Radiation Area.

Which one of the following describes the type of Radiation Work Permit (RWP) and the permission required for TIP Room entry?

A. A Specific RWP with permission given by the RP Lead technician or RP Supervisor only.

B. A Specific RWP with permission given by the RP Manager, or designee, and the on-watch Shift Manager.

C. A General RWP with permission given by the RP Lead technician or RP Supervisor only.

D. A General RWP with permission given by the RP Manager, or designee, and the on-watch Shift Manager.

Proposed Answer: B

Explanation: EN-RP-101 Section 5.6 contains guidance for entry into a Very High Radiation Area (VHRA). This section requires a Specific RWP and approval of the RP Manager, or designee, and the on-watch Shift Manager.

A. Incorrect – VHRA entry requires approval of the RP Manager, or designee, and the on-watch Shift Manager. Entry into a Locked High Rad Area requires approval of the RP Lead technician or RP Supervisor.
C. Incorrect – A Specific RWP is required for VHRA entry. VHRA entry requires approval of the RP Manager, or designee, and the on-watch Shift Manager. Entry into a Locked High Rad Area requires approval of the RP Lead technician or RP Supervisor.
D. Incorrect – A Specific RWP is required for VHRA entry.

Technical Reference(s): EN-RP-101, EN-RP-105

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2010 Audit #71

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.5
	Importance Rating	2.9

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: #73

The Plant is operating at 100% power with the following:

- Movement of radioactive material in the Reactor Building causes Area Radiation Monitor (ARM) #25, RX BLDG EL 272 EAST HCU AREA, to go into alarm high.
- Annunciator 09-3-1-40, RX BLDG ARM RAD HI, is received and acknowledged.

Which one of the following describes how Annunciator 09-3-1-40 will respond to the following two independent conditions?

(1) If a second ARM input exceeds its high alarm setpoint, then Annunciator 09-3-1-40 ____________.

AND

(2) If ARM #25 returns to normal, then Annunciator 09-3-1-40 _____________.

	(1)	(2)
A.	re-flashes	can be reset before depressing ARM RESET pushbutton(s)
B.	re-flashes	can **NOT** be reset before depressing ARM RESET pushbutton(s)
C.	does **NOT** re-flash	can be reset before depressing ARM RESET pushbutton(s)
D.	does **NOT** re-flash	can **NOT** be reset before depressing ARM RESET pushbutton(s)

Proposed Answer: D

Explanation: Annunciator 09-3-1-40 will actuate upon receipt of a high alarm condition from any of 16 Area Radiation Monitors (ARMs). However, once the annunciator is in, it will not re-flash for any subsequent high alarm conditions from any of the other ARMs. Additionally, once the high radiation condition clears, the high alarm trip is sealed in until the RESET pushbutton is depressed on the corresponding ARM trip unit on control room panel 09-11. Until this RESET pushbutton is depressed, both the ARM trip unit amber HIGH light and Annunciator 09-3-1-40 will be sealed in.

A. Incorrect – The annunciator does not have re-flash capability. The annunciator cannot be reset until the ARM RESET pushbutton is depressed.
B. Incorrect – The annunciator does not have re-flash capability.
C. Incorrect – The annunciator cannot be reset until the ARM RESET pushbutton is depressed.

Technical Reference(s): ARP 09-3-1-40, 1.78-98, UFSAR Fig 7.13-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-17 1.14.d.5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:

DK 4/11/12 - Edited question wording to clarify two parts of question are independent conditions, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.50
	Importance Rating	4.2

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question: #74

The Plant is operating at 100% power with the following:

- Annunciator 09-3-1-14, TORUS BULK TEMP HI OR RTD FAILURE, is in alarm
- Torus temperature recorders on Panel 09-3 indicate:
    - TORUS TEMP A 16-1TR-131A: 96^{o}F
    - TORUS TEMP B 16-1TR-131B: 97^{o}F

Which one of the following describes the significance of these indications and the required Operator action?

The annunciator and recorder values are indicative of a…

A. High Torus water temperature condition. Manually scram the Reactor and enter AOP-1, Reactor Scram.

B. High Torus water temperature condition. Enter EOP-4, Primary Containment Control, and initiate Torus cooling.

C. Failed Torus water temperature RTD. At the Relay Room MAP panel, use RTD select switch to identify failed RTD.

D. Failed Torus water temperature RTD. Remove the failed input to annunciator 09-3-1-14 per EN-DC-136, Temporary Modifications.

Proposed Answer: B

Explanation: Annunciator 09-3-1-14 alarms on either high Torus water temperature or failure of a Torus water temperature RTD. Receipt of this alarm requires investigation of Torus water temperature indications to determine cause. The given Torus water temperature indications are above normal and above the EOP-4 entry condition (95^{o}F). Torus water temperature is below the value requiring an immediate manual scram (110^{o}F). Upon entry to EOP-4, Torus cooling is required to restore water temperature below 95^{o}F.

Technical Reference(s): ARP 09-3-1-14, EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11E 4.02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.6
	Importance Rating	3.7

Knowledge of EOP mitigation strategies.

Proposed Question: #75

The Plant was operating at 100% power when a manual scram was inserted due to a steam leak in the Drywell. The following conditions resulted:

- Multiple control rods are stuck at position 48.
- EOP-3, Failure to Scram, is being executed.
- Reactor water level has been intentionally lowered.
- The CRS has directed Reactor water level to be maintained between -19" and 110".

Which one of the following describes the allowed use of injection systems based on location, per EOP-3?

With the exception of SLC, RCIC and CRD, using systems that inject outside the core shroud is ___(1)___ and using systems that inject inside the core shroud is ___(2)___.

	(1)	(2)
A.	allowed	allowed
B.	allowed	**NOT** allowed
C.	**NOT** allowed	allowed
D.	**NOT** allowed	**NOT** allowed

Proposed Answer: B

Explanation: EOP-3, Failure to Scram, contains a Reactor water level mitigation strategy designed to control Reactor power. One of the elements of this strategy is to limit the use of RPV injection sources that inject inside the core shroud. This strategy is implemented through the EOP-3 step that states, “Using only Group 1 Water Level Control Systems, maintain RPV water level between -19 in. and either: (1) Level to which RPV water level was lowered, if it was deliberately lowered, OR (2) 222.5 in., if RPV water level was not deliberately lowered.” Additionally, a separate step in EOP-3 directs, “Terminate and prevent injection from CS until otherwise directed.” Core Spray injects inside the shroud.

A. Incorrect – Core Spray injection (inside shroud) is not allowed in EOP-3.
C. Incorrect – Core Spray injection (inside shroud) is not allowed in EOP-3. Injection outside the core shroud is allowed.
D. Incorrect – Injection outside the core shroud is allowed in EOP-3.

Technical Reference(s): EOP-3, EOP-3a, MIT-301.11D

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11D 1.07

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

DK 4/11/12 - Edited question wording from “LPCI/Core Spray” to “Outside/Inside Shroud” to make general instead of system specific, based on NRC comment.